U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Technical Evaluation Report on the Content of the U.S. Department of Energy's Yucca Mountain Repository License Application

Postclosure Volume: Repository Safety After Permanent Closure

AVAILABILITY PAGE

Technical Evaluation Report on the Content of the U.S. Department of Energy's Yucca Mountain Repository License Application

Postclosure Volume: Repository Safety After Permanent Closure

Manuscript Completed: July 2011

Office of Nuclear Material Safety and Safeguards
Division of High-Level Waste Repository Safety

(Intentionally Left Blank)

ABSTRACT

This "Technical Evaluation Report on the Content of the U.S. Department of Energy's Yucca Mountain License Application; Postclosure Volume: Repository Safety After Permanent Closure" (TER Postclosure Volume) presents information on the NRC staff's review of DOE's Safety Analysis Report (SAR), provided on June 3, 2008, as updated by DOE on February 19, 2009. The NRC staff also reviewed information DOE provided in response to NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, this report provides information on the staff's evaluation of (i) the repository's barriers and (ii) the Total System Performance Assessments (TSPAs) used for the individual protection calculation, the separate groundwater protection calculation, and the human intrusion calculation. DOE submitted information consistent with the Yucca Mountain Review Plan (YMRP).

(Intentionally Left Blank)

CONTENTS[1]

Section Page

[1]In this Technical Evaluation Report (TER), the section numbering used within the volume is based on the Yucca Mountain Review Plan (YMRP). [NRC. 2003. "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: NRC.] The U.S. Nuclear Regulation Commission (NRC) staff used the YMRP to guide its review of information the U.S. Department of Energy (DOE) provided in its Safety Analysis Report (SAR).

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

Section Page

CONTENTS (continued)

Section Page

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

(Intentionally Left Blank)

FIGURES

Figure | Page

TABLES

Table | Page

TABLES (continued)

EXECUTIVE SUMMARY

Background

After docketing the U.S. Department of Energy (DOE) license application seeking a construction authorization for the proposed repository at Yucca Mountain, Nevada, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its review in a Safety Evaluation Report. On March 3, 2010, DOE filed a motion with the Atomic Safety and Licensing Board seeking to withdraw its license application to develop a repository at Yucca Mountain, Nevada. In June 2010, the Board denied the DOE motion. To date, petitions asking the Commission to reverse or uphold this decision are pending before the Commission.

On October 1, 2010, the NRC staff began orderly closure of its Yucca Mountain activities. As part of orderly closure, the NRC staff prepared this technical evaluation report (TER), a knowledge management document. This document captures the NRC staff's technical assessment of information presented in DOE's Safety Analysis Report (SAR), dated June 3, 2008, as amended, and supporting information. The TER describes the NRC staff's technical evaluation of the SAR and, in particular, this TER Postclosure Volume provides technical insights on the application of performance assessment in the context of geologic disposal. The TER was developed using the regulations at 10 CFR Part 63 and guidance in the Yucca Mountain Review Plan (YMRP). The TER does not, however, include conclusions as to whether or not DOE satisfies the Commission's regulations.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. These regulations prescribe requirements governing the licensing (including issuance of a construction authorization) of DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area sited, constructed, or operated at Yucca Mountain, Nevada. Under 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, and a period of operations. The period of operations includes the time during which emplacement would occur, any subsequent period before permanent closure during which the emplaced wastes are retrievable, and permanent closure. In addition, the regulations at 10 CFR Part 63 represent a risk-informed, performance-based (RIPB) approach to the review of geological disposal. The RIPB approach uses risk information to focus the review to areas most significant to safety or performance. Therefore, the TER includes discussions regarding how the staff used risk information in its review.

This technical evaluation report presents information on the NRC staff's assessment of the SAR DOE provided on June 3, 2008, as updated on February 19, 2009.[1] The NRC staff also reviewed information DOE provided in response to NRC staff's requests for additional information, and other information that DOE provided related to the SAR. In conducting its review of DOE's SAR, the NRC staff was guided by the Yucca Mountain Review Plan.[2]

[1]DOE. 2009. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.
[2]NRC. 2003. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC

System Description and Demonstration of Multiple Barriers

A geologic repository is to include multiple barriers, both natural and engineered. Barriers prevent or limit the movement of water or radioactive material and thus isolate waste. A multiple barrier approach ensures that the overall repository system is robust and not wholly dependent on any single barrier to ensure repository safety. DOE is to identify these barriers when it calculates how the repository will perform in the future, describe the capability of each barrier, and provide the technical basis for its description. In its SAR for the proposed repository at Yucca Mountain, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier. Each of these barriers includes multiple features that DOE described as important to waste isolation.

DOE expects that the Upper Natural Barrier will substantially reduce the amount of water that reaches the repository horizon. In semi-arid environments, like that at Yucca Mountain, humidity and precipitation are low and surface evaporation rates are high. In addition, plant uptake and surface runoff can reduce further the amount of water available to move from the surface into the rock layers above the repository. In its SAR, DOE explained that during the first 10,000 years after the repository is closed, most of the water that does reach the depth of the repository is prevented from seeping into the repository and is diverted around waste emplacement tunnels, or drifts, because of thermal effects from the waste. After approximately 10,000 years, DOE concluded that the amount of heat generated from the waste will be low enough to allow water to seep into drifts and potentially contact the EBS.

DOE identified the primary purpose of the EBS as preventing or substantially reducing movement of water that actually contacts the waste, and of limiting movement of dissolved radionuclides away from the repository. DOE predicts that the walls of the waste emplacement drifts will degrade slowly over time. However, DOE expects that specific engineered features will mostly stay in place, will remain largely intact, and will continue to keep the waste substantially dry for very long time periods. If the repository is undisturbed by very large earthquakes or volcanoes, DOE projects that less than 0.01 percent of the waste will be exposed to water that seeps into the drifts during the first 10,000 years after the repository is closed.

In addition to the emplacement drifts, DOE expects other specific features to limit the flow of seepage water or dissolved radionuclides. These features include the drip shields, the waste packages, the waste forms and waste package internal components, and the emplacement pallet and invert. The drip shields divert seepage water away from the waste package. Likewise, as long as the waste packages remain intact, seepage water cannot reach or interact with the enclosed waste forms. Once waste packages begin to corrode, or form cracks, radionuclide releases from the packages are limited by, among other things, the rate at which the various waste forms deteriorate and the rate at which continuous liquid pathways can form through openings large enough to permit flow. DOE projects that most waste packages will remain intact for approximately 200,000 years after the repository is closed, unless large earthquakes or volcanoes occur and cause damage. Once the waste packages begin to leak, DOE expects that the internal components of the waste packages along with the emplacement pallet and invert materials will contribute to a physical and chemical environment around the waste that prevents or substantially reduces the movement of water and dissolved radionuclides away from the repository. DOE projects that the proposed EBS would limit radionuclide releases to the Lower Natural Barrier to less than 0.003 percent of the available inventory at 10,000 years and to not more than 7 percent of the available inventory at 1 million years after the repository is closed.

DOE concluded that the Lower Natural Barrier will impede the movement of most, but not all, radionuclides from the EBS to the accessible environment. Layers of unsaturated rock below the repository 300 m [approximately 1,000 ft] thick are expected to retard the flow of dissolved radionuclides on their way to the water table. Once in saturated rock or groundwater below the repository, water potentially contaminated with radionuclides from the repository must travel through 12–14 km [7–9 mi] of fractured volcanic rock and 4–6 km [2–4 mi] of saturated gravels and sands before reaching the human environment. DOE determined that the effectiveness of the Lower Natural Barrier in retaining specific individual radionuclides depends on the solubility, sorptive properties, and half-life of the specific radionuclide. DOE projects that the releases of solubility-limited, strongly sorbed nuclides are reduced by as much as 99 percent, while those of moderately soluble, low-sorption, long half-life nuclides are reduced by lesser amounts. Highly soluble, nonsorbing radionuclides, however, will not be retained and will move at roughly the same rate as the groundwater flow.

The NRC staff has reviewed the SAR and other information DOE has submitted in support of a system of multiple barriers and determined DOE identified the design features of the EBS and natural barrier features of the geologic setting that are important to isolation of waste at Yucca Mountain. DOE showed that multiple barriers important to waste isolation occur both in the natural and in the engineered systems at the proposed repository at Yucca Mountain. DOE described the capabilities of the barriers it identified and provided a reasonable technical basis for these descriptions. DOE reasonably described the time period during which the barriers perform their intended functions and DOE reasonably accounted for uncertainty in its descriptions of barrier capabilities. DOE's descriptions of barrier capability are consistent with their use in DOE's Total System Performance Assessment (TSPA).

Assessment of Repository Performance

A performance assessment is a systematic analysis that answers three questions: What can happen? How likely is it to happen? and What are the resulting consequences? The NRC staff reviewed the TSPA DOE provided in support of its SAR for the proposed repository at Yucca Mountain. In conducting its review of DOE's TSPA of the repository, the staff evaluated DOE's system description, demonstration of multiple barriers, and associated supporting scientific and analytic methods to focus on those items most important to waste isolation.

To answer the question "What can happen?" after a repository is closed at Yucca Mountain, DOE had to consider a wide range of specific features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, volcanic activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides on rock surfaces), for possible inclusion in (or exclusion from) its evaluation. Once selected, DOE then used these features, events, and processes (FEPs) to postulate a range of credible, future scenarios. A scenario is a well-defined sequence of events and processes, which can be interpreted as an outline of one possible future condition of the repository system. Thus, scenario analysis identifies the possible ways in which the repository environment could evolve to develop a defensible representation of the system and estimate the range of credible potential consequences. After DOE selected appropriate FEPs and used them to postulate scenarios, DOE grouped similar scenarios into scenario classes and screened them for use in its performance assessment of the facility. The NRC staff reviewed DOE's scenario analysis to verify that DOE did not overlook future conditions at the proposed repository that could significantly enhance or degrade its safety. To conduct this review, the staff used its own risk insights from previous performance assessments for the Yucca Mountain site, detailed process-level modeling efforts, laboratory and field experiments, and natural analog studies.

When addressing the question “How likely is it that these events will happen?” DOE assessed the likelihood that credible scenario classes could disrupt repository performance. A performance assessment used for the individual protection calculation considers events that have at least 1 chance in 100 million of occurring. DOE included three disruptive event types for inclusion in its postclosure performance assessments: disruption by volcanic (i.e., igneous) events, disruptions by earthquake (i.e., seismic) events, and early failure of waste packages or drip shields.

To answer the question “What are the resulting consequences?” DOE made estimated projections called “model abstractions” to represent the performance of various parts of the repository system. Each model abstraction develops one or more numerical models that represent how specific FEPs interact and affect performance of repository systems. DOE also included potentially significant variations in site or design characteristics into the models, so that a range of potential outcomes would be calculated in the performance assessment.

To evaluate whether DOE’s model abstractions portrayed the expected consequences when implemented in the overall performance assessment, the NRC staff reviewed 13 separate categories of model abstractions. The NRC staff selected these abstractions from engineered, geosphere, and biosphere subsystems found to be most important to waste isolation on the basis of previous performance assessments, knowledge of site characteristics, and careful examination of DOE’s proposed repository design. In its review, the staff focused on those models of greatest risk significance to repository safety. For the postclosure period, “important to repository safety or waste isolation” means important to estimating release of radionuclides to the accessible environment and annual dose.

For each model abstraction, the NRC staff determined whether the data DOE used appropriately represented site- and design-specific characteristics, including the variability and uncertainty in these characteristics. The NRC staff evaluated how DOE incorporated FEPs in the model abstractions and reviewed DOE’s technical bases to support the inclusion or exclusion of these FEPs. In addition, the NRC staff reviewed the methods DOE used to develop the model abstractions, including how DOE represented model uncertainty. The staff’s review also examined how DOE supported the use of its models in the performance assessment and how DOE considered the potential effects of alternative models.

In reviewing DOE’s SAR and other information submitted, the NRC staff notes that DOE’s performance assessment evaluations included appropriate FEPs and that a reasonable basis was provided for those that were excluded. DOE considered events that have at least 1 chance in 100 million per year of occurring (i.e., igneous events, seismic events, and early failure of waste packages and drip shield events) in its performance assessment analyses. After conducting focused reviews of the 13 model abstractions, the NRC staff notes that those model abstractions are reasonable for use in the performance assessments because (i) applicable data related to natural systems and the EBS were included, (ii) alternative models were appropriately considered, and (iii) appropriate technical bases in support of the model abstractions were provided. DOE should confirm its approach for decay chain radionuclide behavior by providing, through its performance confirmation program, information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone, as identified in TER Section 2.2.1.3.9.3.

Performance Assessment Results

DOE used its TSPA to represent the range of behavior of a repository at Yucca Mountain and to account for uncertainty in FEPs that could affect the evolution of the repository over the postclosure period. DOE developed its analysis of repository performance by grouping scenario classes broadly as either nominal or disruptive. The nominal scenario class comprises those FEPs that are present under “normal” conditions (i.e., eventual infiltration of water, corrosion of waste packages, release of radionuclides, transport of radionuclides in groundwater). During the initial 10,000 years after repository closure, DOE’s nominal scenario class does not result in any dose to the reasonably maximally exposed individual. Disruptive scenario classes, as noted earlier, include additional FEPs that account for the effects of specific events, such as earthquakes and volcanoes, which could disturb or alter the performance of the repository in ways not included under the nominal scenario class.

Disruptive events of sufficient magnitude have the potential to result in doses to the reasonably maximally exposed individual at any time during the postclosure period. The estimate of projected dose resulting from a disruptive event or scenario is to be weighted by the probability that the disruptive event will occur. Therefore, a key component of the NRC staff’s review of DOE’s performance assessment involves a determination that the probabilities and consequences of each of the scenario classes are appropriately included in the average annual dose calculations.

The NRC staff conducted confirmatory calculations to supplement and assist its review of DOE’s TSPA. The confirmatory calculations provide both a quantitative understanding of the assessment and an understanding of whether there is a general consistency between submodels of the analysis and the overall results, including uncertainty. For example, the staff’s confirmatory calculations examined whether DOE’s projections of the timing and extent of breaching of the waste packages corresponded and were consistent with the projected timing and magnitude of the average annual dose. The staff’s confirmatory calculations were performed for selected time periods (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide the staff a perspective on the time-dependent nature of waste package failure, associated radioactive decay and release of specific radionuclides. In assessing the credibility of DOE’s average annual dose curve for the groundwater pathway, the NRC staff conducted separate confirmatory calculations for (i) the amount of water entering failed waste packages, (ii) the release of radionuclides from the waste packages, (iii) transport of radionuclides through the unsaturated and unsaturated zones, (iv) effects of disruptive events, and (v) annual dose to the reasonably maximally exposed individual.

The staff’s confirmatory calculations were based on the NRC staff’s review of DOE’s TSPA calculation, including DOE’s models and intermediate outputs. Thus NRC staff’s confirmatory calculations address key quantitative attributes of the repository system to help evaluate overall performance. This approach provided the staff with a straightforward method for confirming that DOE’s TSPA results provided a credible representation of the repository’s performance. In other words, NRC staff’s results showed that DOE’s average annual dose curve is consistent with the model abstractions, probabilities, and treatment of uncertainties, each of which were reviewed separately using the NRC staff’s guidance in the Yucca Mountain Review Plan.

DOE presented the overall average annual dose curve due to releases from the repository over the entire 1-million-year period in its SAR. The peak of the curve of overall average annual dose to the RMEI is approximately 0.003 mSv [0.3 mrem] per year over the initial

10,000 years and is approximately 0.02 mSv [2 mrem] per year over the 1-million-year period after repository closure.

DOE estimated the concentrations for the alpha activity (including background) would be 0.5 pCi/L with the largest contribution coming from natural background radiation already in the groundwater. The largest annual release from the repository of relevant alpha-emitting radionuclides into the representative volume from the repository was estimated to be more than 1,000 times less than background levels.

DOE estimated the dose from beta- and photon-emitting radionuclides would be 0.0006 mSv [0.06 mrem] per year for the whole body and the largest dose to any organ would be 0.0026 mSv [0.26 mrem] per year as a result of drinking 2 L [0.53 gal] of water per day assumed to be at peak estimated concentration levels of radionuclides released from the repository in the representative volume.

DOE estimated the concentration from the combined Ra-226 and Ra-228 (including background) would be 0.5 pCi/L with the largest contribution coming from natural background radiation already in the groundwater. The largest annual release from the repository of Ra-226 and Ra-228 into the representative volume from the repository was estimated to be almost 1 million times less than the background levels.

DOE determined the earliest time after disposal that waste packages would have degraded sufficiently such than an intrusion could occur without a driller recognizing it, as part of the human intrusion analysis. DOE selected 200,000 years as a conservative assumption of the earliest time the waste packages could degrade enough so that an intrusion could occur without drillers recognizing it. DOE developed a separate performance assessment to evaluate the consequences of a postulated human intrusion event assumed to occur 200,000 years after permanent closure of the repository. DOE modified its performance assessment for individual protection to represent the human intrusion scenario. DOE’s estimated dose due to releases from the repository is approximately 0.0001 mSv [0.01 mrem] per year shortly after the time of the intrusion.

Summary Conclusions

The NRC staff has reviewed the SAR and other information DOE submitted in support of its SAR. DOE submitted information consistent with the guidance in the YMRP. Specifically, NRC staff notes that the repository (i) is composed of multiple barriers; (ii) the Total Systems Performance Assessments (TSPAs) used for the individual protection, human intrusion, and separate groundwater protection calculations are reasonable; and (iii) the technical approach and results in DOE’s TSPA, including the average annual dose values and the performance of the repository barriers, discussed in this TER, are reasonable. DOE should confirm its approach for decay chain radionuclide behavior by providing, through its performance confirmation program, information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone, as identified in TER Section 2.2.1.3.9.3.

ACRONYMS AND ABBREVIATIONS

AFM	active fracture model
AMR	analysis and model reports
APE	annual probability of exceedance
ASTM	American Society for Testing and Materials
BDCF	biosphere dose conversion factors
BSC	Bechtel SAIC Company, LLC.
BWR	boiling water reactor
CDSP	codisposal packages
CFR	Code of Federal Regulations
CFu	Crater Flat undifferentiated
CHn	Calico Hills nonwelded
CNWRA®	Center for Nuclear Waste Regulatory Analyses
CRWMS M&O	Civilian Radioactive Waste Management Management & Operation
CSNF	commercial spent nuclear fuel
DHLW	defense high-level waste
DOE	U.S. Department of Energy
DVRGFSM	Death Valley Regional Groundwater Flow System Model
EBS	engineered barrier system
EPA	U.S. Environmental Protection Agency
ERMYN	Environmental Radiation Model for Yucca Mountain Nevada
FAR	Fortymile Wash Ash Redistribution
FEP	feature, event, and process
GROA	geologic repository operations area
ITWI	important to waste isolation
MASSIF	Mass Accounting System for Soil Infiltration and Flow
MCO	multicanister overpack
MDEB	mechanical disruption of engineered barrier
MIC	microbially influenced corrosion
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NC–EWDP	Nye County Early Warning Drilling Program
OECD	Organisation for Economic Co-operation and Development
PFDHA	probabilistic fault displacement hazard analysis
PGA	peak ground acceleration
PGV	peak ground velocity
PSHA	probabilistic seismic hazard assessment
PTn	Paintbrush Tuff nonwelded
PVHA	probabilistic volcanic hazard assessment
PVHA-U	probabilistic volcanic hazard assessment-update
PWR	pressurized water reactor
RAI	request for additional information
RB	repository block
RMEI	reasonably maximally exposed individual
RMS	root-mean-square
RST	residual stress threshold
SA	spectral accelerations
SAR	Safety Analysis Report

ACRONYMS AND ABBREVIATIONS (continued)

SCC	stress corrosion cracks
SDFR	slip-dissolution aging and film-rupture
SEF	sorption enhancement factor
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SSHAC	Senior Seismic Hazard Analysis Committee
SZEE	saturated zone flow and transport expert elicitation
TAD	transportation, aging, and disposal
TEDE	total effective dose equivalent
TER	Technical Evaluation Report
TSPA	Total System Performance Assessment
TSw	Topopah Spring welded
UDEC	universal distinct element code
USGS	U.S. Geological Survey
UZ	unsaturated zone
WRIP	water-rock interaction parameter
YMRP	Yucca Mountain Review Plan

INTRODUCTION

After docketing the U.S. Department of Energy (DOE) license application seeking a construction authorization for the proposed repository at Yucca Mountain, Nevada, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its review in a Safety Evaluation Report. On March 3, 2010, DOE filed a motion with the Atomic Safety and Licensing Board seeking to withdraw its license application to develop a repository at Yucca Mountain, Nevada. In June 2010, the Board denied the DOE motion. To date, petitions asking the Commission to reverse or uphold this decision are pending before the Commission.

On October 1, 2010, the NRC staff began orderly closure of its Yucca Mountain activities. As part of orderly closure, the NRC staff prepared this technical evaluation report (TER), a knowledge management document. This document captures the NRC staff's technical assessment of information presented in DOE's Safety Analysis Report (SAR), dated June 3, 2008, as amended, and supporting information. The TER describes the NRC staff's technical evaluation of the DOE SAR and, in particular, this TER Postclosure Volume provides technical insights on the application of performance assessment in the context of geologic disposal. The TER was developed using the regulations at 10 CFR Part 63 and guidance in the Yucca Mountain Review Plan (YMRP). The TER does not, however, include conclusions as to whether or not DOE satisfies the Commission's regulations.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. These regulations prescribe requirements governing the licensing (including issuance of a construction authorization) of DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area sited, constructed, or operated at Yucca Mountain, Nevada. Under 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, and a period of operations. The period of operations includes the time during which emplacement would occur, any subsequent period before permanent closure during which the emplaced wastes are retrievable, and permanent closure. In addition, the regulations at 10 CFR Part 63 represent a risk-informed, performance-based (RIPB) approach to the review of geological disposal. The RIPB approach uses risk information to focus the review to areas most significant to safety or performance. Therefore, the TER includes discussions regarding how the staff used risk information in its review. In conducting its review, the NRC staff was guided by the YMRP.

This TER Postclosure Volume documents the results of the NRC staff's assessment of repository performance after a repository is permanently closed. The NRC staff's technical evaluation considered (i) the geologic repository's multiple barriers, both natural and manmade, or engineered, and (ii) the performance assessments, including model abstractions, used for the individual protection calculation, the separate groundwater protection calculation, and the human intrusion calculation.

Risk-Informed and Performance-Based Review

The NRC staff evaluated DOE's performance assessment using a risk-informed and performance-based review. DOE's performance assessment is a systematic analysis that answers three basic questions that often are used to define risk: What can happen? How likely is it to happen? and What are the resulting consequences? The Yucca Mountain performance assessment is a sophisticated analysis that involves various complex considerations and evaluations. Examples include evolution of the natural environment; degradation of engineered barriers; and disruptive events, such as seismicity and igneous activity. Because the performance assessment encompasses such a broad range of issues, the NRC staff used risk information throughout the review process to ensure that the review focused on those items most important to waste isolation. YMRP Section 2.2.1 provides guidance to the NRC staff to apply risk information throughout the review of the performance assessment.

To support its risk-informed, performance-based review, the NRC staff initially reviewed DOE's information on the repository's barriers. Important barriers (engineered and natural) of the performance assessment are identified, each barrier's capability is described, and the technical basis for that capability is provided in the SAR. This risk information describes DOE's understanding of each barrier's capability to prevent or substantially delay the movement of water or radioactive materials. The NRC staff's review of DOE information regarding the repository's barriers provides an understanding of each barrier's importance to waste isolation, which helps focus the NRC staff's review of DOE's Total System Performance Assessment (TSPA) presented in TER Sections 2.2.1.2, "Scenario Analysis and Event Probability"; 2.2.1.3, "Model Abstraction"; and 2.2.1.4, "TSPA Calculations." Particular parts of the NRC staff's review are emphasized on the basis of the risk insights (i.e., those attributes of the repository system most important to repository performance). Additionally, the NRC staff has considered independent risk insights from previous performance assessments conducted for the Yucca Mountain site, detailed process modeling efforts, laboratory and field experiments, and natural analog studies, and has identified this information, as appropriate.

System Description and Demonstration of Multiple Barriers

A geologic repository at Yucca Mountain is to include multiple barriers, both natural and engineered. Barriers prevent or limit the movement of water or radioactive material. A multiple barrier approach ensures that the overall repository system is robust and not wholly dependent on any single barrier to ensure repository safety. DOE is to identify these barriers when it calculates how the repository will perform in the future, describe the capability of each barrier and provide the technical basis for its description. In its SAR for the proposed repository at Yucca Mountain, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier. The Upper Natural Barrier is composed of features above the repository (i.e., topography, surficial soils, and the unsaturated zone) that reduce the movement of water downward toward the repository, which in turn reduces the rate of movement of water from the radioactive waste in the repository to the accessible environment. The engineered barrier system includes different engineering features (e.g., emplacement drifts, drip shields, waste packages and their internal components, and emplacement pallets and inverts) that are designed to (i) enhance the performance of the waste package, preventing radionuclide releases while it is intact; (ii) limit radionuclide releases after the waste package is breached by limiting the amount of water that can contact the waste package; and (iii) limit radionuclide release from the engineered barrier system through sorption processes. The Lower Natural Barrier is composed of features below the repository (i.e., unsaturated zone) and from the repository location to the boundary of the accessible environment and the RMEI location (i.e., saturated zone) that reduce the rate of radionuclide movement from the repository to the accessible environment through such processes as the slow movement of water and sorption of radionuclides onto mineral surfaces. Each of these barriers includes features that DOE described as important to waste isolation. The NRC staff's review is provided in TER Section 2.2.1.1.

Review of Postclosure Total System Performance Assessment

A performance assessment is a systematic analysis that answers the three questions that define risk: What can happen? How likely is it to happen? and What are the resulting consequences? The NRC staff reviewed the Total System Performance Assessment (TSPA) analytic models and analyses DOE provided in support of its SAR.

Scenario Analysis and Event Probability

To answer the question, "What can happen?" after a repository is closed, DOE considered a wide range of specific features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, volcanic activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides on rock surfaces) for possible inclusion in (or exclusion from) its TSPA model. Once specific features, events, and processes were selected for inclusion, DOE then used these features, events, and processes (FEPs) to postulate a range of credible, future scenarios. A scenario is a well-defined sequence of events and processes, which can be interpreted as an outline of one possible future condition of the repository system. Thus, scenario analysis identifies the possible ways in which the repository environment could evolve so that a defensible representation of the system can be developed to estimate the range of credible potential consequences. After the features, events, and processes are selected and used to postulate scenarios, similar scenarios are grouped into scenario classes, which are screened for use in the performance assessment model. The goal of scenario analysis is to ensure that no important aspect of the potential high-level waste repository is overlooked in the evaluation of its safety.

Consistent with this general guidance and the review areas in YMRP Section 2.2.1.2.1, the NRC staff evaluates DOE's scenario analysis in four separate TER sections (Sections 2.2.1.2.1.3.1 to 2.2.1.2.3.4). Section 2.2.1.2.1.3.1 contains the NRC staff's evaluation of both DOE's methodology to develop a list of features, events, and processes and DOE's list of the features, events and processes that it considered for inclusion in the performance assessment analyses. In Section 2.2.1.2.1.3.2, the NRC staff evaluates DOE's screening of its list of features, events and processes, including DOE's technical bases for the exclusion of features, events, and processes from its performance assessment. DOE's formation of scenario classes and the exclusion of specific scenario classes in DOE's performance assessment analyses are evaluated in Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4, respectively.

The NRC staff's evaluation of DOE's methodology and conclusions on the probability of events included in the performance assessments is addressed in TER Section 2.2.1.2.2. That chapter is aimed at the second risk triplet question: How likely is it to happen?

Model Abstraction

The NRC staff's evaluation of DOE's model abstractions focuses on the consequences of overall repository performance. In particular, the NRC staff's evaluation considers the model abstractions used in DOE's TSPA to represent the performance (i.e., expected annual doses) of the repository.

The review of the model abstraction process begins with the review of the repository design and the data characterizing the geology and the performance of the design and proceeds through the development of models used in the performance assessment. The model abstraction review process ends with a review of how the abstracted models are implemented in the TSPA model (e.g., parameter ranges and distributions, integration with model abstractions for other parts of the repository system, representation of spatial and temporal scales, and whether the performance assessment model appropriately implements the abstracted model). For example, the review of parameter distributions considers the relevant data, the corresponding uncertainty, and effects on repository performance (i.e., the dose to the reasonably maximally exposed individual). The potential for risk dilution—the lowering of the risk, or dose, from an unsupported parameter range and distribution—is also part of this model abstraction review.

In many applications, a conservative approach can be used to decrease the need to collect additional information or to justify a simplified modeling approach. A conservative approach may overestimate the dose to the reasonably maximally exposed individual. Approaches designed to overestimate a specific aspect of repository performance (e.g., higher temperatures within the drifts) may be conservative with respect to temperature but could lead to nonconservative results with respect to dose. The TSPA is a complex analysis with many parameters, and DOE may use conservative assumptions to simplify its approaches and data collection needs. However, a technical basis that supports the selection of models and parameter ranges or distributions is provided. The NRC staff's evaluation of the technical bases supporting models and parameter ranges or distributions considers whether the approach results in calculated doses that would overestimate, rather than underestimate, the dose to the reasonably maximally exposed individual. In particular, the claim of conservatism as a basis for simplifying models and parameters is evaluated to verify that any simplifications do not unintentionally result in nonconservative results.

The intentional use of conservatism to manage uncertainty also has implications for the NRC staff's efforts to risk inform its review. The NRC staff evaluates assertions that a given model or parameter distribution is conservative from the perspective of overall system performance (i.e., the dose to the reasonably maximally exposed individual). The NRC staff used any available information to risk inform its review. For example, if DOE used an approach that overestimates a specific aspect of repository performance, then the NRC staff would consider the effects of this approach on other parts of the TSPA model, overall repository performance, and the representation or sensitivity of important phenomena.

The NRC staff has separated the model abstraction review into the following 13 categories that are addressed in TER Sections 2.2.1.3.1 through 2.2.1.3.14.[1]

1. Degradation of Engineered Barriers (TER Section 2.2.1.3.1)

   This TER section provides the NRC staff's evaluation of the chemical degradation of the drip shields and waste packages emplaced in the repository drifts. Chemical degradation is primarily associated with the effect of corrosion processes on the metal surfaces of the drip shields and waste packages. The drip shields and the waste packages are engineered barriers, a subset of the engineered barrier system. The general functions of the engineered barrier system are to (i) prevent or significantly

[1]It was decided to discuss two of the topics in the YMRP as a single topic in the TER; however, the numbering system in the YMRP was retained in the TER to the maximum extent possible (e.g., Biosphere Characteristics is Section 2.2.1.3.14 in the YMRP and the TER). The review of Airborne Transport of Radionuclides (YMRP Section 2.2.1.3.11) and Redistribution of Radionuclides in Soil (YMRP Section 2.2.1.3.13) is discussed in TER Section 2.2.1.3.13 because the NRC staff considers a single discussion of these two topics provides for more clarity in the TER. Thus, the TER does not contain a section numbered 2.2.1.3.11.

reduce the amount of water that contacts the waste, (ii) prevent or significantly reduce the rate at which radionuclides are released from the waste, and (iii) prevent or significantly reduce the rate at which radionuclides are released from the engineered barrier system to the Lower Natural Barrier. The complete engineered barrier system consists of the emplacement drifts, the drip shields, the waste packages, the naval spent nuclear fuel structure, the waste forms and waste package internal components, and emplacement pallets and inverts.

2. Mechanical Disruption of Engineered Barriers (TER Section 2.2.1.3.2)

This TER section provides the NRC staff's evaluation of the mechanical disruption of the drip shields and waste packages emplaced in the repository drifts. Mechanical disruption of engineered barrier system components generally results from external loads generated by accumulating rock rubble. Rubble accumulation can result from processes such as (i) degrading emplacement drifts due to thermal loads, (ii) time-dependent natural weakening of rocks, and (iii) effects of seismic events (vibratory ground motion or fault displacements). During seismic events, rubble loads on engineered barrier system components can increase as the accumulated rock rubble is shaken.

3. Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms (TER Section 2.2.1.3.3)

This TER section provides the NRC staff's evaluation of (i) the chemistry of water entering the drifts, (ii) the chemistry of water in the drifts (tunnels), and (iii) the quantity of water in contact with the engineered barrier system. These three abstraction topics provide input to model the features and performance of the engineered barrier system (e.g., drip shields and waste packages). For example, in its SAR, DOE relied on corrosion tests that were conducted on waste package and drip shield materials under a range of geochemical environments. The range of testing environments was derived from a range of potential starting water compositions and from knowledge of near-field and in-drift processes that alter these compositions.

4. Radionuclide Release Rates and Solubility Limits (TER Section 2.2.1.3.4)

This TER section provides the NRC staff's evaluation of the processes that could result in water transport of radionuclides out of the engineered barrier system, including the waste packages and the emplacement inverts, and into the unsaturated zone (the rock mass directly below the repository horizon and above the water table). The engineered barrier system and the transport pathway within the drift (repository tunnel) are the initial barriers to radionuclide release. If a waste package is breached and water enters the waste package, the radionuclides contained in the package may be released from the engineered barrier system.

5. Climate and Infiltration (TER Section 2.2.1.3.5)

This TER section provides the NRC staff's evaluation of the representation of climate and infiltration. This evaluation considers the reduction of water flux from precipitation to net infiltration. Because of the generally vertical movement of percolating water through the unsaturated zone in DOE's representation of the natural system, water entering the unsaturated zone at the ground surface (infiltration) is the only source for deep percolation water in the unsaturated zone at and below the proposed repository.

6. Unsaturated Zone Flow (TER Section 2.2.1.3.6)

This TER section provides the NRC staff's evaluation of the abstraction of flow in that portion of the repository system above the water table (i.e., the unsaturated zone). Water percolating through the unsaturated zone above the repository (i.e., Upper Natural Barrier) may enter drifts, providing the means to interact with and potentially corrode the waste packages. Water percolating through the unsaturated zone below the repository (i.e., Lower Natural Barrier) also provides a flow pathway for transporting radionuclides downward to the water table. Once radionuclides pass below the water table, they may subsequently move laterally within the saturated zone to the accessible environment.

7. Radionuclide Transport in the Unsaturated Zone (TER Section 2.2.1.3.7)

This TER section provides the NRC staff's evaluation of the representation of radionuclide transport in the unsaturated zone. The NRC staff's evaluation focuses on (i) advection, because most of the radionuclide mass is carried through the unsaturated zone by water flowing downwards to the water table; (ii) sorption, because sorption in porous media in the southern half of the repository area has the largest overall effect on slowing radionuclide transport in the unsaturated zone; (iii) matrix diffusion in fractured rock, because matrix diffusion coupled with sorption slows radionuclide transport in the northern half of the repository area; (iv) colloid-associated transport, because radionuclides attached to colloids may travel relatively unimpeded through the unsaturated zone; and (v) radioactive decay and ingrowth, because these processes affect the quantities of radionuclides released from the unsaturated zone over time.

8. Flow Paths in the Saturated Zone (TER Section 2.2.1.3.8)

This TER section provides the NRC staff's evaluation of the representation of flow paths in the saturated zone (i.e., the direction and magnitude of water movement in the saturated zone). Flow paths in the saturated zone provide the pathway for releases of radionuclides to migrate from the saturated zone below the repository to the accessible environment {approximately 18 km [11 mi] south of the repository}. The magnitude (specific discharge) of water flow is used to determine the speed that water moves through the saturated zone.

9. Radionuclide Transport in the Saturated Zone (TER Section 2.2.1.3.9)

This TER section provides the NRC staff's evaluation of transport of radionuclides in the saturated zone. The NRC staff's technical review focuses on (i) how DOE represented the geological, hydrological, and geochemical features of the saturated zone in a framework for modeling the transport processes; (ii) how DOE integrated the saturated zone transport abstraction with other TSPA abstractions for performance assessment calculations; and (iii) how DOE included and supported important transport processes in the saturated zone radionuclide transport abstraction.

10. Igneous Disruption of Waste Packages (TER Section 2.2.1.3.10)

This TER section provides the NRC staff's evaluation of models for the potential consequences of disruptive igneous activity at Yucca Mountain if basaltic magma rising through the Earth's crust intersects and enters a repository drift or drifts (DOE's igneous intrusion modeling case) or enters a drift and later erupts to the surface through one or more conduits (DOE's volcanic eruption modeling case). The proposed Yucca Mountain repository site lies in a region that has experienced sporadic volcanic events in the past few million years, such that DOE previously determined the probability of future igneous activity at the site to exceed 1×10^{-8} per year. The NRC staff's technical review evaluates subsurface igneous processes (i.e., intrusion of magma into repository drifts, waste package damage, and formation of conduits to the surface), which involves entrainment of waste into the conduit and toward the surface. These processes control the amount of radionuclides that can be released during a potential igneous event.

11. Concentration of Radionuclides in Groundwater (TER Section 2.2.1.3.12)

This TER section provides the NRC staff's evaluation of the concentration of radionuclides in groundwater extracted by pumping and used in the annual water demand. Radionuclides transported through the saturated zone via groundwater to the accessible environment may be available for extraction by a pumping well. The reasonably maximally exposed individual is assumed to use well water with average concentrations of radionuclides and has an annual water demand of 3.7×10^{9} L [3,000 acre-ft].

12. Airborne Transport and Redistribution of Radionuclides (TER Section 2.2.1.3.13)

This TER section reflects the NRC staff's evaluation of the volcanic ash exposure scenario and the groundwater exposure scenario. First, this TER section provides the NRC staff's evaluation of the airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption and the subsequent redistribution of those radionuclides in soil. Second, this TER section evaluates redistribution of radionuclides in soil that arrive in the accessible environment through groundwater transport. This section addresses both airborne transport of radionuclides (YMRP Section 2.2.1.3.11) and redistribution of radionuclides in soil (YMRP Section 2.2.1.3.13).

13. Biosphere Characteristics (TER Section 2.2.1.3.14)

This TER section provides the NRC staff's evaluation of the model used to calculate biosphere transport and the annual dose to the reasonably maximally exposed individual. The biosphere model calculates the transport of radionuclides within the biosphere through a variety of exposure pathways (e.g., soil, food, water, air) and applies dosimetry modeling to convert the reasonably, maximally exposed individual exposures into annual dose. Exposure pathways in the biosphere model are based on assumptions about residential and agricultural uses of the water and indoor and outdoor activities. These pathways include ingestion, inhalation, and direct exposure to radionuclides deposited to soil from irrigation. Ingestion pathways include drinking contaminated water, eating crops irrigated with contaminated water, eating food products produced from livestock raised on contaminated feed and water, eating farmed fish raised in contaminated water, and inadvertently ingesting soil. Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products.

Total System Performance Assessment (TSPA) Calculations

DOE has conducted an analysis, through its TSPA, that evaluates repository behavior in terms of groundwater concentrations and annual dose due to potential releases from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a Yucca Mountain repository. In reviewing DOE's postclosure performance calculations (i.e., individual protection, human intrusion, and separate limits for protection of groundwater), the NRC staff's review evaluates (i) the performance assessment scenario classes (a set or combination of features, events and processes that are used in the performance assessment to represent a class or type of scenario, such as seismic activity), (ii) the representation of the scenario classes within the performance assessment (e.g., the performance assessment results are consistent with the models, parameters, and assumptions that make up the performance assessment), and (iii) the statistical stability of the annual dose that the performance assessment calculates.

The NRC staff has separated its review of DOE's TSPA calculations in TER Sections 2.2.1.4.1 through 2.2.1.4.3:

1. Postclosure Individual Protection Calculation (TER Section 2.2.1.4.1)

   This TER section provides the NRC staff's review of the individual protection calculation. The performance assessment used for the individual protection calculation considers both likely and unlikely events and all radiological exposure pathways.

2. Human Intrusion Calculation (TER Section 2.2.1.4.2)

   This TER section provides the NRC staff's review of the human intrusion calculation (i.e., exploratory drilling for groundwater results in a borehole penetrating a waste package in the repository—the timing of the intrusion is set to the earliest time after disposal that the waste packages would degrade sufficiently that a human intrusion could occur without the drillers recognizing it). The performance assessment used for the human intrusion calculation considers likely events and all radiological exposure pathways.

3. Separate Groundwater Protection Calculation (TER Section 2.2.1.4.3)

   This TER section provides the NRC staff's review of the separate groundwater protection calculation. The performance assessment used for the separate groundwater protection calculation considers likely events and the drinking water exposure pathway.

Expert Elicitation (TER Section 2.5.4)

This TER section provides the NRC staff's review of the three expert elicitations DOE used in support of its SAR. Expert elicitations were conducted in the areas of seismic hazard (SAR Section 2.2.2.1), igneous activity (SAR Section 1.1.6.2, Section 2.2.2.2, and Section 2.3.11), and saturated zone flow and transport (SAR Section 2.3.9.2).

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts. Expert judgments are routinely used to evaluate and interpret the factual bases of SARs. The NRC staff has acknowledged that DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, DOE used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information such as data collection, analyses, and experimentation. In this context, the NRC staff has reviewed DOE’s use of expert elicitation, which includes a technical review of the results of these elicitations.

(Intentionally Left Blank)

CHAPTER 1

2.2.1.1 System Description and Demonstration of Multiple Barriers

2.2.1.1.1 Introduction

This chapter of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) description of the capabilities of the barriers in the geologic repository. The provisions in 10 CFR Part 63 for the repository after permanent closure require that the geologic repository include multiple barriers, consisting of both natural barriers and an engineered barrier system. Natural and engineered barriers isolate waste by preventing or substantially reducing the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment. A comprehensive description of the capabilities of the natural and engineered barriers would identify the risk-significant attributes for the repository performance. The technical basis for the barrier capability is evaluated in TER Section 2.2.1.3.

A system of multiple barriers is intended to ensure that the repository system is robust and is not wholly dependent on a single barrier. Such a system is more robust in handling failures and external challenges. Therefore, 10 CFR 63.113 requires that a geologic repository contain both natural and engineered barriers.

The emphasis of the NRC staff's integrated review of DOE's performance assessment calculation is not solely focused on the isolated performance of individual barriers, but rather on ensuring that the repository system is robust. The purpose of this chapter is to provide an understanding of how the natural barriers and the engineered barrier system work in combination to enhance the resiliency of the geologic repository. Therefore, to increase the NRC staff's understanding of integrated repository performance, 10 CFR 63.115 requires that DOE

- Identify the important barriers of the performance assessment evaluation
- Describe each barrier's capability
- Provide a technical basis for that capability which is based on and consistent with the performance assessment used to evaluate repository performance

The description of barrier capability provides risk information that helps the NRC staff interpret the performance assessment results by describing how different elements of the performance assessment affect the overall performance of the system. This understanding can guide the NRC staff in the review of the technical bases for those aspects that are important, thereby allowing increased confidence in postclosure performance. The NRC staff can use this risk information to implement a risk-informed approach in its review of DOE's performance assessment calculations.

The NRC staff's evaluation is based on information provided in the Safety Analysis Report (SAR) (DOE, 2008ab), as supplemented by DOE responses to the NRC staff's requests for additional information (DOE, 2009an,bu). DOE provided a description of the barrier capabilities in SAR Chapter 2.1. This description, supplemented by DOE's responses to the NRC staff's requests for additional information, is used by the NRC staff in its review of the technical bases for the performance assessment, as documented in TER Section 2.2.1.3, which focuses on DOE's description of the barrier capabilities. As discussed in the staff's Yucca Mountain Review Plan (YMRP) Section 2.2.1 (NRC, 2003aa), the multiple barrier review focuses on each barrier's importance to waste isolation.

2.2.1.1.2 Evaluation Criteria

NRC staff's review of multiple barriers is guided by criteria in 10 CFR 63.113(a) and 10 CFR 63.115(a–c), which require the following:

- The geologic repository to include multiple barriers, consisting of both natural and engineered barriers

- Identification of those features of the repository that are considered barriers important to waste isolation (ITWI)

- A description of the capabilities of those barriers

- A technical basis for the description of the capability, which is based on and consistent with the technical basis for the performance assessment

Definitions and discussions of important terms and concepts, such as "barrier" and "important to waste isolation," are located in 10 CFR 63.2 and 10 CFR 63.102(h). For example, 10 CFR 63.2 states that the term "barrier" means any material, structure, or feature that, for a period to be determined by NRC, prevents or substantially reduces the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or prevents the release or substantially reduces the release rate of radionuclides from the waste.
The NRC staff reviewed information provided in the Safety Analysis Report (SAR) (DOE, 2008ab), as supplemented by DOE responses to NRC staff's requests for additional information (DOE, 2009an,bu), using the review methods and acceptance criteria provided in YMRP Section 2.2.1.1. The three YMRP acceptance criteria that provide guidance for the NRC staff's review of multiple barriers under 10 CFR 63.115 are

- Identification of barriers is adequate
- Description of barrier capability to isolate waste is acceptable
- Technical basis for barrier capability is adequately presented

The following technical evaluation is largely organized according to these three acceptance criteria. Because the description of the barrier capability and the technical basis for the barrier capability are interrelated, the review activities associated with these two acceptance criteria are discussed together in TER Section 2.2.1.1.3.2.

2.2.1.1.3 Technical Evaluation

Summary of DOE SAR

DOE identified the barriers considered important to waste isolation and summarized their capabilities and technical bases in SAR Section 2.1. This summary relies on more extensive information documented in SAR Sections 2.2 through 2.4. DOE documented the analyses that it used to identify and evaluate barrier capability in SNL (2008ad). As described in that document, DOE's identification of the barriers, and the description of the capability of these barriers, is based on the scenario analysis summarized in SAR Section 2.2 that identifies and evaluates the features, events, and processes to be considered in the Total System Performance Assessment (TSPA) model. The TSPA is an abstracted model that quantitatively integrates inputs from the various supporting analytic models. DOE used this abstracted integrating model to analyze repository performance.

DOE identified three barriers (upper natural, engineered, and lower natural) and provided the features or components that make up these barriers in SAR Section 2.1.1.

DOE summarized the capability of the barriers in SAR Section 2.1.2. For each barrier, DOE identified and provided a brief qualitative description of the key processes and events that influence the capability of each barrier. DOE provided some of these descriptions at the level of individual barrier features or components (e.g., the waste package component of the engineered barrier system). DOE provided other descriptions at an aggregate level (e.g., the waste form, waste canisters, and waste package internals taken together). DOE then described

- The specific function of each barrier component and how the barrier component carries out its functions

- The time period over which the barrier functions and how DOE expects the capability of the barrier to evolve over time

- How uncertainty in the barrier capability has been accounted for in the performance assessment

- The impact of disruptive events on the barrier, if any

- A quantitative evaluation of the barrier capability to carry out its barrier functions

DOE summarized the technical basis for the description of barrier capability in SAR Section 2.1.3. DOE stated that the technical basis for the barrier capability is the same as the technical basis for the model used in the TSPA analyses. DOE also stated that the technical basis for the description of the barrier capability is provided in SAR Section 2.3. SAR Table 2.1-5 identified which TSPA model abstractions are associated with each barrier; SAR Section 2.1.3 identified the location of the technical basis for the description of the barrier capability for those abstractions. SAR Section 2.1.3 briefly summarized the technical basis for each TSPA model component. Each summary identified the subsection of SAR Section 2.3 where DOE described the technical basis of the model component in more detail.

2.2.1.1.3.1 Identification of Barriers

The NRC staff reviewed DOE's discussion of how it identified barriers important to waste isolation in SAR Section 2.1.1. DOE identified three barriers (upper natural, engineered, and lower natural) and then provided the features or components that made up these barriers. In SAR Table 2.1-1, DOE identified the safety classification (i.e., whether DOE considers the feature or component to be important to waste isolation) of each feature or component. These barriers include features that are important to waste isolation from the upper and lower natural barrier and components that are important to waste isolation from the engineered barrier system. In DOE Enclosure 1 (2009an), DOE expanded SAR Table 2.1-1 to include

- The features, events, and processes considered important to barrier capability
- A qualitative discussion of how the stated barrier functions are attained
- For each barrier feature or component considered important to waste isolation, a quantitative summary of barrier capability based on information from the performance assessment analysis

Because DOE's safety classification identified in SAR Table 2.1-1 indicates individually whether each feature or component is considered important to waste isolation and is linked to its capability and to either the upper natural barrier, engineered barrier system, or the lower natural barrier, the NRC staff notes that DOE has identified the barriers that are relied on for repository performance. Because this list includes features from both the engineered and natural systems, DOE identified barriers that include at least one feature from the engineered system and one from the natural system.

DOE identified three engineered system components as important to waste isolation based solely on their capability to reduce the probability of criticality.

2.2.1.1.3.2 Description and Technical Basis for Barrier Capability

NRC Staff's Review Process

The NRC staff's review of the description and technical basis for barrier capability is based on a list of 22 individual features presented in SAR Table 2.1-1. For purposes of evaluation, the NRC staff consolidated these features to yield nine features as shown in Table 1-1. The NRC staff consolidated these 22 features by noting that several features appeared to be related in the second column of Table 2.1-1. For example, the emplacement drift is referred to twice. The NRC staff, therefore, consolidated the two emplacement drift entries into a grouped entry titled "Emplacement Drift." Also, 11 of the features were prefaced with the term "Waste Form and Waste Package Internals." The NRC staff grouped all nine of the features into one component titled "Waste Form and Waste Package Internals." The NRC staff included cladding into this

Table 1-1. Summary of Staff's Barrier Component Review

Barrier	Barrier Feature	SAR Table 2.1-1 ITWI Components	SAR Table 2.1-1 Non-ITWI Components
Upper Natural Barrier	Topography and Surficial Soils	Topography and Surficial Soils	None
Upper Natural Barrier	Unsaturated Zone Above the Repository	Unsaturated Zone Above the Repository	None
Engineered Barrier System (EBS)	Emplacement Drift	Emplacement Drift	Emplacement Drift: Nonemplacement Openings, Closure, Ground Support, and Ventilation System

Table 1-1. Summary of Staff's Barrier Component Review (continued)			
Barrier	**Barrier Feature**	**SAR Table 2.1-1 ITWI Components**	**SAR Table 2.1-1 Non-ITWI Components**
EBS	Waste Form and Waste Package Internals	• Transport, aging, and disposal (TAD) canister • Naval canister • Commercial spent nuclear fuel (SNF) and high-level waste glass • Naval SNF • Naval SNF canister system components* • TAD canister internals* • DOE SNF canister internals*	• DOE SNF canister • High-level waste canister • Codisposal package internals • DOE SNF • Cladding
EBS	Emplacement Pallet and Invert	None	• Waste Package Pallet • Invert
Lower Natural Barrier	Unsaturated Zone Below the Repository	Unsaturated Zone Below the Repository	None
Lower Natural Barrier	Saturated Zone	Saturated Zone	None
*DOE identified these components as important to waste isolation solely in relation to their capability to reduce the probability of criticality.			

grouped category by noting that cladding is a component that contains the waste form and is internal to the package. The NRC staff also noted that neither the emplacement pallet nor the invert were classified as important to waste isolation, and that both components serve to support the waste package. The NRC staff, therefore, consolidated these two features into a single feature. The resulting consolidated list is consistent with the grouping that DOE used in SAR Section 2.1.2.2 in its summary of the features, processes, and characteristics of the engineered barrier system that are important to waste isolation.

The NRC staff reviewed the descriptions of the barrier capability of these nine consolidated features. To evaluate the description of the barrier capability, the NRC staff reviewed how DOE

- Identified the safety classification and primary function of each barrier component

- Identified the characteristics and processes important to barrier capability, including both those that are potentially beneficial and those that are potentially harmful to barrier functions

- Described how the barrier component was represented in the performance assessment

- Described the qualitative and quantitative capabilities of each barrier component, consistent with the performance assessment analyses

- Characterized the time period over which the barrier functions and how DOE expects the barrier capability to change over time

- Accounted for the uncertainty in the description of the barrier capability

To evaluate the technical basis for the barrier capability, the NRC staff reviewed the consistency between the descriptions of the barrier capability documented in SAR Section 2.1.2 and the technical bases summarized in SAR Section 2.1.3 and further documented in SAR Section 2.3. In addition, the NRC staff reviewed the description of the performance confirmation plan to determine whether it was consistent with the descriptions of barrier capability. TER Section 2.4 contains the results of the NRC staff's review of the performance confirmation plan.

The NRC staff also considered the insights gained from NRC Appendix D (2005aa), as updated (CNWRA and NRC, 2008aa), to determine whether DOE had omitted any features or processes that might contribute significantly to barrier capability in its description of barrier capability. In addition, the NRC staff reviewed DOE's TSPA model described in SNL (2008ag) to assess consistency between the descriptions of barrier capability and how the TSPA model components actually represented the barrier capability.

The NRC staff summarizes the results of the review of the individual barrier components, as identified in the second column of TER Table 1-1. In each of the following sections, the NRC staff's evaluation

- Describes whether the barrier capability is explained in terms of a capability to prevent or substantially reduce the rate of movement of water or radionuclides

- Identifies the SAR sections where DOE described the capability of each barrier component and briefly summarizes the described capabilities

- Describes whether the identified capabilities are consistent with the results from the Total System Performance Assessment; in reviewing these analyses, the NRC staff examined whether the numerical results, used to illustrate barrier capability, were consistent with the intermediate results used to compute the dose in the Total System Performance Assessment calculation

- Identifies where DOE has described the time period over which the barrier performs its stated function and briefly summarizes whether DOE has described the time period over which the barrier performs its stated function

- Identifies where DOE has described the uncertainty in the barrier capability and describes whether DOE accounted for uncertainties in its characterization and modeling of the barriers

- Identifies where DOE summarized the technical basis for barrier capability, describes whether this technical basis is consistent with the technical basis for the performance assessment models, and describes whether the technical basis is commensurate with the importance of each barrier's capability

2.2.1.1.3.2.1 Upper Natural Barrier: Topography and Surficial Soils

The NRC staff reviewed DOE's description of the barrier capability of the topography and surface soils. DOE described the barrier capabilities of the topography and the surficial soils qualitatively in SAR Section 2.1.2.1.1 and quantitatively in SAR Section 2.1.2.1.6.1. DOE supplemented this description in DOE Enclosure 1 (2009an). DOE used net infiltration as a percentage of annual precipitation to quantify the barrier capability of topography and surficial soils. The NRC staff notes that the information provided by DOE describes a capability that substantially reduces infiltration into the unsaturated zone, which in turn reduces the rate of movement of water from the nuclear waste in the repository to the accessible environment.

DOE's climate and infiltration analyses are summarized in SAR Tables 2.3.1-2, 2.3.1-3, and 2.3.1-4. DOE stated in SAR Section 2.1.2.1.1 and in DOE Enclosure 1 (2009an) that for approximately 10,000 years following closure of the repository, a limited amount of water would infiltrate into the unsaturated zone above the repository at Yucca Mountain. DOE attributed the low rate of infiltration to low precipitation that is substantially further reduced by high rates of evapotranspiration (e.g., uptake by plants, surface evaporation) and surface runoff. In SAR Section 2.1.2.1.1, DOE stated that the average net infiltration rate estimates range from about 3 to 17 percent of the total precipitation, depending upon the climate state and the infiltration scenario. For the post-10,000-year period, DOE stated that it used the deep percolation rate in the proposed 10 CFR Part 63. In DOE Enclosure 6 (2009cb), DOE stated that use of the distribution of deep percolation specified in the final 10 CFR Part 63 led to an insignificant increase in dose.

On the basis of its review of the DOE description of the barrier capabilities of topography and surficial soils in SAR Section 2.1.2.1 and in DOE Enclosure 1 (2009an), the NRC staff notes that DOE's description of the barrier capability is consistent with the results from the performance assessment calculation because DOE based this description on intermediate results from its infiltration model used in the performance assessment, as documented in SAR Section 2.3.1.

DOE provided information in SAR Section 2.1.2.1.3 on the time period over which this upper natural barrier feature performs its intended function. DOE stated that the topography and surficial soils are not expected to change significantly in the 10,000 years following closure, but changes in climate and vegetation are expected to affect the barrier capability during this period. In SAR Tables 2.3.1-17 to 2.3.1-19 and DOE Enclosure 1 (2009an), DOE addressed when it expects different climate states to occur and provided infiltration rates under different climate scenarios. Because DOE explicitly discussed the time dependence of the infiltration rate, DOE has reasonably described the time period over which the topography and surface soils perform as a barrier.

In SAR Section 2.1.2.1.4, DOE described sources of uncertainty that are considered in the climate and infiltration model. Sources of uncertainty include (i) the interpretation of the geologic record of past climates, (ii) the parameters describing evapotranspiration, (iii) the applicability of models, and (iv) the characteristics of the Yucca Mountain site. DOE also addressed the uncertainty in the barrier capability by describing results from its infiltration model demonstrating the probability of different infiltration scenarios (SAR Section 2.3.2.4.1.2.4.5 and Table 2.3.2-27). DOE described in its Enclosure 5 (2009bo) that adjusting the probability weighting of these scenarios based on deep subsurface observations of chloride and temperature (SAR 2.3.2.4.1.2.4.5 and Table 2.3.2-27) reduced the average infiltration fluxes in the initial 10,000 years following permanent closure by approximately 50 percent. DOE addressed infiltration uncertainties in the post-10,000-year period in SAR Section 2.3.2.4.1.2.4.2

by using a weighting of net infiltration scenarios to yield a distribution of deep percolation fluxes comparable to the distribution specified in the proposed 10 CFR Part 63. DOE Enclosure 6 (2009cb) stated that use of the distribution of deep percolation specified in the final 10 CFR Part 63 led to an insignificant increase in dose. Because DOE described the sources of uncertainty and presented the range of uncertainty in its infiltration estimates, DOE accounted for uncertainty in its descriptions of barrier capability.

In SAR Section 2.1.3.1, DOE summarized the technical basis of the barrier capability description of the upper natural barrier, which includes the topography and surficial soils component. In this discussion, DOE indicated which TSPA models it used to support the description of the barrier capability of the upper natural barrier. DOE based its description of the barrier capability of the topography and surficial soils on the climate and infiltration model described in SAR Section 2.3.1.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.1.1 and 2.1.3.1 with SAR Section 2.3.1 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.1.1 and 2.1.3.1 with the results of the climate and infiltration model described in SAR Section 2.3.1 and notes that the net infiltration calculation results are consistently represented among these SAR sections. The NRC staff therefore notes that the technical basis for the description of barrier capability in SAR Sections 2.1.2.1.1 and 2.1.2.1.6.1 is consistent with the technical basis of the climate and infiltration model.

TER Section 2.2.1.3.5 documents the NRC staff's evaluation of the infiltration model that provides the technical basis for this capability. The NRC staff notes in TER Section 2.2.1.3.5 that DOE provided reasonable technical bases for the climate and infiltration model and for the range of net infiltration values used in the performance assessment calculations. Therefore, the NRC staff notes that DOE's technical basis for the description of the barrier capability of the topography and surface soils is commensurate with the barrier capability described in SAR Section 2.1.2.1.1, SAR Section 2.1.2.1.6.1, and in DOE Enclosure 1 (2009an).

In summary, the NRC staff notes that the capability of the topography and surficial soils to prevent or substantially reduce the rate of movement of water from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis for the performance assessment.

2.2.1.1.3.2.2 Upper Natural Barrier: Unsaturated Zone Above the Repository

The NRC staff reviewed the DOE description of the barrier capability of the unsaturated zone above the repository. DOE described the capability of the unsaturated zone above the repository to prevent or substantially reduce seepage qualitatively in SAR Section 2.1.2.1.2 and quantitatively in SAR Section 2.1.2.1.6.2. DOE supplemented this description in DOE Enclosures 1 and 2 (2009an). The information provided by DOE describes a capability to prevent or substantially reduce seepage of water into the emplacement drifts, which in turn substantially reduces the rate of water movement from the nuclear waste in the repository to the accessible environment.

In SAR Section 2.1.2.1.6.2, DOE explained that the average percolation flux at the repository depth is, at most, a few percent less than the average net infiltration near the surface above the repository. In other words, changes in the flow rate of water between the ground surface and the repository level are relatively small, indicating to the NRC staff that DOE did not credit any significant processes resulting in the diversion of water away from the emplacement drift location. However, DOE explained in SAR Section 2.1.2.1.2, SAR Section 2.1.2.1.6.2, and in DOE Enclosures 1 and 2 (2009an) that capillary diversion of water at the host rock–air interface at the drift wall prevents much of the water flowing in the rock at the repository level from entering the drift as seepage (i.e., dripping). DOE explained that at some drift locations, all of the water is diverted around the drift, resulting in no drips at all; at others, only some of the water enters, and the remainder is diverted around the drift. In addition, the short duration, relatively higher flow rates resulting from infiltration following brief episodes of precipitation are spread out in time and space as they pass through the Paintbrush Tuff. In DOE Enclosure 2 (2009an), DOE explained that this damping of episodic infiltration pulses by the Paintbrush Tuff results in water flow rates below the Paintbrush Tuff that are consistently lower than the peak flow rate during the infiltration pulse, but which are more nearly constant over time (i.e., steady-state fluxes below the Paintbrush Tuff). DOE explained that because capillary diversion processes are more effective at low percolation flow rates, the damping of episodic infiltration pulses by the Paintbrush Tuff contributes to the effectiveness of the capillary barrier. DOE quantified the barrier capability of the unsaturated zone above the repository for each of the five percolation subregions for the climate states projected for the first 10,000 years after repository closure (SAR Section 2.1.2.1.2). DOE used an analysis based on the TSPA seepage models and inputs to show that average seepage rates range from less than 1 to about 17 percent of the percolation fluxes for intact drifts within the first 10,000 years following closure, as described in DOE Enclosure 3, Table 11 (2009bo). DOE expects capillary forces to divert more than 80 percent of percolation flux away from the intact drifts for the initial 10,000 years after closure. DOE Enclosure 3, Table 5 (2009bo) identified that for intact drifts, the fraction of the repository experiencing dripping conditions (i.e., the seepage fraction) ranges from 10 to 70 percent. Results for the collapsed drift case, which is a likely scenario in the post-10,000-year period, are shown in DOE Enclosure 3, Table 11 (2009bo). These results indicate that mean seepage percentage ranges from about 40 to 56 percent in the post-10,000-year period. DOE expects that capillary forces would divert at least 44 percent of percolation flux away from collapsed drifts. The post-10,000-year seepage fractions for the corresponding flow fields range from about 44 to 89 percent, as described in DOE Enclosure 3, Table 8 (2009bo).

On the basis of its review of the DOE description of the barrier capabilities of the unsaturated zone above the repository in SAR Section 2.1.2.1, the NRC staff notes that DOE's description of the barrier capability is consistent with the results from the performance assessment calculation because DOE refers to analyses in SAR Section 2.3.3.4.2 and in DOE Enclosure 3 (2009bo) that are based on TSPA models and input data.

In SAR Section 2.1.2.1.3, DOE provided information on the time period over which this upper natural barrier feature performs its intended function. DOE stated that the unsaturated zone above the repository is not expected to change in the 10,000 years following closure and that changes in the barrier capability are due to changes in infiltration. SAR Figure 2.1-5 describes how seepage changes as a function of time. DOE has reasonably described the time period over which the unsaturated zone above the repository performs as a barrier because DOE explicitly considered how the ability of the drift to divert water changes over time.

In SAR Section 2.1.2.1.4, DOE discussed sources of uncertainty in the barrier capability of the unsaturated zone above the repository. These primarily are associated with uncertainties in the models and the characteristics of the Yucca Mountain site. SAR Tables 2.1-6 through 2.1-9 provide the range of uncertainty in seepage fractions. DOE also discussed these uncertainties in SAR Section 2.3.3.4.2. Because DOE described sources of uncertainty and explained how these uncertainties affect the rate and extent of seepage, the NRC staff notes that DOE has considered uncertainty in its descriptions of barrier capability.

SAR Section 2.1.3.1 summarized the technical basis of the barrier capability of the upper natural barrier, which includes the unsaturated zone above the repository. In its discussion, DOE indicated which TSPA models it used to support the description of the barrier capability of the upper natural barrier. DOE based its description of the barrier capability of the unsaturated zone above the repository on the unsaturated zone flow model described in SAR Section 2.3.2 and on the seepage (ambient and thermal) models described in SAR Section 2.3.3.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.1.2 and 2.1.3.1 with SAR Sections 2.3.2 and 2.3.3 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.1.6.2 and 2.1.3.1 with the results of the site-scale unsaturated zone flow and seepage models described in SAR Sections 2.3.2 and 2.3.3 and notes that the deep percolation, seepage, and seepage fraction estimates are consistently represented among these SAR sections. The technical bases for the description of barrier capability in SAR Sections 2.1.2.1.2 and 2.1.2.1.6.2, as supplemented in DOE Enclosures 1 and 2 (2009an), are consistent with the technical bases of the site-scale unsaturated zone flow and seepage models.

TER Section 2.2.1.3.6 documents the NRC staff's evaluation of the technical basis for the unsaturated zone flow and seepage model abstractions that form the basis for this capability. The NRC staff notes in TER Section 2.2.1.3.6 that DOE provided technical bases for the site-scale unsaturated zone flow and seepage models and for the ranges of deep percolation, seepage, and seepage fraction values used in the performance assessment that address their intended use. DOE's technical bases for the description of the barrier capability of the unsaturated zone above the repository are commensurate with the barrier capability described qualitatively in SAR Section 2.1.2.1.2, quantitatively in SAR Section 2.1.2.1.6.2, and as supplemented by DOE Enclosures 1 and 2 (2009an).

In summary, the NRC staff notes that (i) the capability of the unsaturated zone above the repository to prevent or substantially reduce the rate of movement of water from the Yucca Mountain repository to the accessible environment is reasonably described and (ii) the technical basis for the barrier capability is based on and is consistent with the technical basis for the performance assessment.

2.2.1.1.3.2.3 Engineered Barrier System: Emplacement Drift

The NRC staff reviewed the DOE description of the barrier capability of the emplacement drift. DOE discussed the barrier capabilities of the emplacement drift in SAR Section 2.1.2.2 under the discussion titled "Emplacement Drift" and in DOE Enclosures 1 and 3 (2009an). DOE stated that the capability of the emplacement drift to prevent or substantially reduce the movement of water is associated with the capillary barrier discussed under the upper natural barrier. DOE associated the capability of the drift to prevent or reduce the rate of movement of radionuclides with the effect of temperature and water chemistry on various processes affecting the degradation of the other EBS components (e.g., drip shield, waste package, and waste form) and radionuclide transport. DOE Enclosures 1 and 3 (2009an) specifically identified and discussed the roles of individual processes in the capability of the emplacement drifts.
The NRC staff notes that DOE has described the emplacement drifts capabilities to include

- The intact emplacement drift opening represents a zero-capillarity feature within the rock formation that supports diversion of unsaturated zone flow around the opening, which reduces the rate of seepage into the drift

- The collapsed, rubble-filled emplacement drift provides reduced seepage diversion capabilities and limits drip shield and waste packages motion under seismic activity

- The mechanical integrity of the drift provides a stable environment that controls the mechanical and chemical degradation of the drip shield and waste package, which divert seepage water and prevent or limit the rate of contact of water with the waste form

- The mechanical integrity of the drift provides a stable environment that controls the rate of waste form degradation and the chemical conditions within the waste package, which control the rate of movement of radionuclides

This information shows DOE has reasonably described the capabilities of the emplacement drift with respect to drift seepage by describing the effect of an intact and collapsed drift on the performance of the capillary barrier associated with the unsaturated zone above the repository. DOE has reasonably described the capabilities of the emplacement drift with respect to the effect of the in-drift environment because DOE described the effect of the in-drift environment on degradation and transport processes within the drift that are described as aspects of the other components of the engineered barrier system.

DOE discussed the time period over which the emplacement drift functions in SAR Section 2.1.2.2.3 and in DOE Enclosure 3 (2009an). DOE described the evolution of the mechanical stability of the drift and the in-drift environment and discussed how these changes affect the major processes associated with emplacement drift performance. These evaluations, along with the discussion of the time period over which DOE expects the drift to degrade due to seismic events, provide a reasonable understanding of the time period over which the emplacement drift performs its function because DOE described both the timing and effect of seismically induced drift degradation. In TER Section 2.2.1.3.2.3.2, the NRC staff addresses the capabilities of the emplacement drift related to the mechanical integrity of the drift opening.

DOE discussed the uncertainty in the performance of the emplacement drift in SAR Sections 2.1.2.2.4 and 2.3.4.4.8 and in DOE Enclosure 3 (2009an). DOE indicated that the uncertainties in the environmental conditions are a primary source of uncertainty in the performance of the engineered barrier system. In SAR Section 2.3.4.4.8, DOE discussed the sources and treatment of uncertainty in the evaluation of rockfall and presented the effect of these uncertainties in SAR Figure 2.1-14, which provides the range of uncertainties in the expected fraction of the drift filled with rubble. The NRC staff determines that these discussions, supplemented by probabilistic outputs of the rockfall model showing the range of times for rubble to accumulate within the drift, show how DOE has accounted for uncertainty in its descriptions of barrier capability.

DOE summarized the technical basis of the engineered barrier system capability, which included the emplacement drift, in SAR Section 2.1.3.2. In its discussion, DOE indicated which TSPA models it used to support the description of the barrier capability of the engineered barrier system. DOE based its description of the barrier capability of the emplacement drift on three TSPA submodels: (i) the ambient and thermal seepage models described in SAR

Section 2.3.3, (ii) the engineered barrier system mechanical degradation model described in SAR Section 2.3.4, and (iii) the in-drift chemical and physical environment model described in SAR Section 2.3.5.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.2 and 2.1.3.2 with SAR Sections 2.3.3, 2.3.4, and 2.3.5 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.2 and 2.1.3.2 with the results of the three emplacement drift TSPA submodels DOE described in SAR Sections 2.3.3, 2.3.4, and 2.3.5 and notes that the seepage rate estimates, the expected time of collapse of the drifts, and the in-drift physical and chemical environment are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Section 2.1.2.2 are consistent with the technical basis of the emplacement drift TSPA submodels.

In TER Sections 2.2.1.3.2, 2.2.1.3.3, and 2.2.1.3.6, the NRC staff evaluates the technical bases used to support the barrier capability of the emplacement drift. The NRC staff notes in TER Sections 2.2.1.3.2, 2.2.1.3.3, and 2.2.1.3.6 that DOE provided technical bases for the emplacement drift TSPA submodels and for the range of values for the seepage rates and the expected time of collapse of the drifts, and the in-drift physical and chemical environment used in the performance assessment, that address their intended use. Therefore, DOE's technical bases for the description of the barrier capability of the emplacement drift are commensurate with the barrier capability described in SAR Section 2.1.2.2 and in DOE Enclosures 1 and 3 (2009an).

In summary, the NRC staff notes that the capability of the emplacement drift to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical bases for the barrier capability are based on and consistent with the technical bases for the performance assessment.

2.2.1.1.3.2.4 Engineered Barrier System: Drip Shield

The NRC staff reviewed the description of the barrier capability of the drip shield. DOE discussed the capability of the drip shield to prevent or substantially reduce contact of seepage with the waste package in SAR Section 2.1.2.2 under the discussion titled "Drip Shield," in SAR Section 2.1.2.2.1, and quantitatively in SAR Section 2.1.2.2.6. DOE supplemented its discussion in DOE Enclosure 1 (2009an). DOE addressed the drip shield's capability to prevent seepage water from contacting the waste package during the thermal period in SAR Section 2.1.2.2. During the thermal period, seepage water, if contacting the waste package, could lead to water chemistry that may initiate localized corrosion. The NRC staff notes that the information provided by DOE describes the capability to prevent or substantially reduce the rate of movement of water.

DOE does not expect extensive drip shield failures before 100,000 years. General corrosion of the drip shield enhances the vulnerability of the drip shield to seismic events as the drip shield plates become thinner as a result of corrosion. DOE expects drip shields to fail between 200,000 and 300,000 years, when general corrosion has weakened the drip shield plates sufficiently such that a seismic event can rupture them. DOE attributes the capability of the drip shield to divert water to corrosion-resistant materials coupled with a low probability of mechanical damage from seismic events and a relatively benign chemical environment during the thermal period.

On the basis of its review of the DOE description of the barrier capabilities of the drip shield in SAR Section 2.1.2.2, SAR Section 2.1.2.2.6, and DOE Enclosure 1 (2009an), the NRC staff notes that DOE's description of the barrier capability is consistent with the results from the performance assessment calculation because DOE described the capability using intermediate results from the TSPA showing the distribution of drip shield failure times.

SAR Section 2.1.2.2.3 addressed the period over which the engineered barrier system, including the drip shield, performs its barrier function. DOE stated that the barrier capability of the drip shield and waste package is not impacted until sufficient corrosion has occurred to create breaches in the waste package. SAR Section 2.1.2.2.6 quantified the change in the effectiveness of the capability of the drip shield. DOE reasonably described the period over which the barrier performs its function because it described how the drip shields degrade over time, and DOE supplemented its description with time-dependent outputs from the drip shield degradation model.

DOE described the sources of uncertainty in the drip shield capability in SAR Section 2.1.2.2.4 and quantitatively described the effect of uncertainties in SAR Section 2.1.2.2.6. These include, for example, uncertainties in the environmental conditions affecting the drip shield. DOE described specific analyses of uncertainty in the model abstractions for drip shield degradation in SAR Sections 2.3.4.5 and 2.3.6.8. The NRC staff notes that these discussions, supplemented by probabilistic outputs of the drip shield degradation model showing the range of times for failure of the drip shield, show how DOE accounted for uncertainty in its descriptions of barrier capability.

SAR Section 2.1.3.2 summarized the technical basis of the barrier capability of the engineered barrier system, which includes the drip shields. In its discussion, DOE identified which TSPA models it used to support the description of the barrier capability of the engineered barrier system. DOE based its description of the barrier capability of the drip shield on three TSPA submodels: (i) the mechanical damage model described in SAR Section 2.3.4.5, (ii) the general corrosion model described in SAR Section 2.3.6.8.1, and (iii) the early failure model described in SAR Section 2.3.6.8.4.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.2 and 2.1.3.2 with SAR Sections 2.3.4.5, 2.3.6.8.1, and 2.3.6.8.4 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.2 and 2.1.3.2 with the results of the three drip shield TSPA submodels DOE described in SAR Sections 2.3.4.5, 2.3.6.8.1, and 2.3.6.8.4 and notes that the drip shield failure time estimates are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Section 2.1.2.2 are consistent with the technical basis of the drip shield TSPA submodels.

In TER Sections 2.2.1.3.1 and 2.2.1.3.2, the NRC staff evaluates the technical bases used to support the barrier capabilities of the drip shield. The NRC staff notes in TER Sections 2.2.1.3.1 and 2.2.1.3.2 that DOE provided technical bases for the drip shield TSPA submodels and for the range of values for the drip shield failure time used in the performance assessment that address their intended use. Therefore, DOE's technical bases for the description of the barrier capability of the drip shields are commensurate with the barrier capability described in SAR Section 2.1.2.2 and in DOE Enclosure 1 (2009an).

In summary, the NRC staff notes that the capability of the drip shield to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis for the performance assessment.

2.2.1.1.3.2.5 Engineered Barrier System: Waste Packages

The NRC staff reviewed the description of the barrier capability of the waste packages. DOE discussed the capability of the waste package to prevent or substantially reduce contact of seepage with the waste form in SAR Section 2.1.2.2 under the discussion titled “Waste Package,” in SAR Sections 2.1.2.2.1 and 2.1.2.2.2, and quantitatively in SAR Section 2.1.2.2.6. DOE Enclosures 1 and 4 (2009an) supplemented this discussion. The capabilities DOE described identify a capability of the barrier to substantially reduce the movement of water or radionuclides.

DOE also credited the barrier capability of the waste package for radionuclide transport through the engineered system. Specifically, the waste package inner vessel contains a large amount of stainless steel that corrodes after breach of the outer vessel. The corrosion products contain high sorption capabilities for some radionuclides. Although the important to waste isolation component that DOE credits for this capability is the waste package inner vessel, DOE described the barrier capabilities associated with sorption to corrosion products as an aspect of the waste form and waste package internal components. TER Section 2.2.1.1.3.2.6 addresses the NRC staff’s evaluation of this barrier capability.

In SAR Section 2.1.2.2, DOE attributed the capability of the waste package to divert water to corrosion-resistant materials coupled with a low probability of mechanical damage from seismic events and a relatively benign chemical environment. DOE discussed the incidence of waste package failure and concluded that extensive early failures of the waste packages are unlikely. DOE does not expect extensive waste package failures to occur until a seismic event capable of damaging the waste packages occurs. Although a model for localized corrosion is included in the TSPA analysis, DOE expects that the presence of the drip shields over the entire thermal period will prevent the occurrence of localized corrosion under the nominal or seismic scenarios. DOE indicated that waste package failures before approximately 200,000 years are primarily due to seismically induced stress corrosion cracking of codisposal waste packages containing DOE standard canisters and high-level waste. DOE attributed the higher resilience of the commercial spent nuclear fuel waste packages under seismic conditions, relative to the codisposal packages, to the damping provided by the massive transport, aging, and disposal canister containing the commercial spent fuel. Upon failure of the drip shield and filling of the drift with rubble, damage from further seismic events is unlikely. Subsequent failures are largely associated with nominal processes affecting both commercial spent nuclear fuel waste packages and codisposal waste packages. Under nominal conditions, DOE expects approximately 50 percent of both the commercial spent nuclear fuel and codisposal waste packages to fail by stress corrosion cracking by 1 million years. The earliest general corrosion waste package failure (at the 95th percentile) is predicted to occur at 560,000 years. At 1 million years, about 10 percent of the waste packages are predicted to fail from general corrosion.

The ability of a breached waste package to prevent or reduce water flow is dependent upon the type and extent of the failure. DOE modeled stress corrosion crack breaches as allowing only diffusive release from the waste package. Larger breaches (primarily due to general corrosion and rarely due to rupture or puncture of the waste package during a seismic event) allow water flow, but a small breached area may limit the rate at which water may enter the waste package. DOE Enclosure 4 (2009an), based on the flux-splitting submodel documented in SAR Section 2.3.7.12.3.1, indicated that a waste package breached by general corrosion is still capable of significant water diversion provided that the breach area is limited to a small percentage of the waste package surface area.

On the basis of its review of the DOE description of the barrier capabilities of the waste packages in SAR Section 2.1.2.2, the NRC staff notes that DOE's description of the barrier capability is consistent with the results from the performance assessment calculation. In particular, the barrier capabilities are (i) described by reference to intermediate results from the performance assessment showing waste package failure times and breached areas and (ii) supported by analyses based on TSPA models and parameters showing water flow rates through breached packages.

SAR Section 2.1.2.2.3 addressed the period over which the engineered barrier system, including the waste package, performs its barrier function. DOE stated that the barrier capability of the drip shield and waste package is not impacted until sufficient corrosion has occurred to create breaches in the waste package. SAR Section 2.1.2.2.6 described the change in the effectiveness of the capability by providing time-dependent outputs of the waste package degradation model. Because DOE has described how the capability degrades over time, and has provided intermediate outputs demonstrating how and when DOE expects the waste packages to fail, DOE has reasonably described the time period over which the waste package performs its barrier function.

SAR Section 2.1.2.2.4 described the sources of uncertainty in the modeled performance of the engineered barrier system. These include, for example, uncertainties in the environmental conditions affecting the waste package, in the temperature dependence of the general corrosion rate, in the effect of microbially induced corrosion, and in the treatment of stress corrosion cracking and localized corrosion. DOE Enclosure 4 (2009an) also addresses the effects of uncertainty in corrosion processes on the ability of the waste package to divert water, noting that in most realizations, there is no breach of any waste package. Because DOE has described the sources of uncertainty, described how these uncertainties are addressed in the TSPA, and provided probabilistic outputs demonstrating the range of uncertainty in waste package failure times and breached area, DOE has provided information that shows it has accounted for uncertainty in its description of the barrier capability of the waste package.

DOE summarized the technical basis of the barrier capability description of the engineered barrier system, which includes the waste packages, in SAR Section 2.1.3.2. DOE based its description of barrier capability of the waste package on six TSPA submodels: (i) the early failure model described in SAR Section 2.3.6.6.3, (ii) the general corrosion model described in SAR Section 2.3.6.3.3, (iii) the localized corrosion model described in SAR Section 2.3.6.4.3, (iv) the stress corrosion cracking model described in SAR Section 2.3.6.5.3, (v) the mechanical damage model described in SAR Section 2.3.4.5, and (vi) the flux-splitting model described in SAR Section 2.3.7.12.3.1.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.2 and 2.1.3.2 with SAR Sections 2.3.6.6.3, 2.3.6.3.3, 2.3.6.4.3, 2.3.6.5.3, 2.3.4.5, and 2.3.7.12.3.1 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections

2.1.2.2 and 2.1.3.2 with the results of the six waste package TSPA submodels DOE described in SAR Sections 2.3.6.6.3, 2.3.6.3.3, 2.3.6.4.3, 2.3.6.5.3, 2.3.4.5, and 2.3.7.12.3.1 and notes that the waste package lifetime, waste package failed area, and waste package water flow estimates are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Section 2.1.2.2 are consistent with the technical bases of the waste package TSPA submodels.

TER Sections 2.2.1.3.1, 2.2.1.3.2, and 2.2.1.3.3 address the NRC staff's evaluation of the technical bases used to support the barrier capability of the waste packages. The NRC staff notes in TER Sections 2.2.1.3.1, 2.2.1.3.2, and 2.2.1.3.3 that DOE provided technical bases for the waste package TSPA submodels and for the ranges of values for the waste package lifetime, waste package failed area, and waste package water flow used in the performance assessment that address their intended use. Therefore, DOE's technical bases for the description of the barrier capability of the waste packages are commensurate with the barrier capability described previously.

In summary, the NRC staff notes that the capability of the waste package to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis used in the performance assessment calculation.

2.2.1.1.3.2.6 Engineered Barrier System: Waste Form and Waste Package Internal Components

The NRC staff reviewed the description of the barrier capability of the waste form and waste package internal components. DOE provided a qualitative description of how the waste form and waste package internal components limit the release of radionuclides from a failed waste package in SAR Section 2.1.2.2 under the discussion titled "Waste Form and Waste Package Internals," in SAR Section 2.1.2.2.2, and in DOE Enclosures 1 and 5 (2009an). DOE described the performance of the waste package internal components quantitatively in SAR Section 2.1.2.2.6. DOE discussed the impacts of these processes in an aggregated fashion, using a metric that indicates the extent to which radionuclides are retained within the entire engineered system over time. Specifically, the approach identifies, for selected radionuclides, the decayed cumulative release from the engineered system (i.e., the amount of radionuclides existing within either the lower natural barrier or the accessible environment relative to the total inventory in the entire system). DOE described that the barrier is capable of substantially reducing the rate of movement of radionuclides from the waste package.

DOE attributed the barrier capability of the waste form and waste package internal components to a number of significant processes that can affect release rates. These processes include waste form degradation, precipitation and dissolution, colloid generation and stability, and sorption to and desorption from waste package internal components. These processes are in turn affected by the chemistry of the aqueous solution inside the failed waste packages as well as the water flow rate within the package. DOE described how these processes limit releases from the engineered barrier system and how these processes are associated with the different internal components of the waste package in DOE Enclosures 1 and 5 (2009an). NRC staff concluded that DOE provided the following information:

- DOE Enclosure 1 (2009an) provided a discussion of the relationship between specific processes and the safety classification of individual waste form and waste package internal components. For example, DOE explained that it considers the waste package inner vessel to be the dominant source of corrosion products for corrosion product sorption, and that corrosion of other internal components is not as significant and is therefore not considered important to waste isolation.

- DOE provided specific information on the rate at which waste forms degrade in DOE Enclosure 5, Section 1.1 (2009an). DOE provided calculations on the basis of TSPA input parameters that evaluate mean waste form lifetimes on the order of up to a few thousand years for spent fuel and tens to hundreds of thousands of years for high level waste glass waste forms, as identified in DOE Enclosure 5, Tables 1.1-1 and 1.1-2 (2009an). On the basis of the DOE results provided in DOE Enclosure 5, Tables 1.1-1 and 1.1-2 (2009an), the high-level waste glass waste form lifetime is significantly more uncertain, with waste form lifetimes that can range anywhere from a few hundred years to over 100 million years.

- DOE indicated in DOE Enclosure 5, Section 1.2 (2009an), on the basis of a selection of TSPA realizations, the effectiveness of the limited breach area associated with cracks for retaining radionuclides under diffusive release conditions and showed that the effect of breach area on release is nuclide and breach area dependent. DOE concluded that for soluble nuclides, releases are sensitive to the breach area for low breach area fractions, but become insensitive to the breach area as the breached area approaches just a few hundred square millimeters. For sorbing, solublility-limited nuclides, DOE concluded that the sensitivity persists for higher breached areas.

- DOE Enclosure 5, Sections 1.3 and 1.4 (2009an) indicated the effectiveness of solubility limits and sorption to corrosion products for limiting the releases from the waste package, on the basis of sensitivity analyses and selected realizations. DOE observed that for the relatively insoluble, sorbing nuclides such as Np-237 and Pu-242, both precipitation/dissolution processes and sorption onto corrosion products are significant in limiting releases from the engineered barrier system.

- DOE addressed the significance of colloidal processes in DOE Enclosure 5, Section 1.5 (2009an). DOE does not identify transport facilitated by colloidal suspensions as significant to the barrier capability of the waste form and waste package internal components. DOE explained that colloids do not facilitate significant releases relative to dissolved forms of the same radionuclides.

On the basis of its review of the information DOE presented in SAR Section 2.1.2.2 and in DOE Enclosures 1 and 5 (2009an), the NRC staff notes that the capabilities of the waste form and waste package internal components described by DOE and summarized in the preceding paragraphs are consistent with the results from the performance assessment calculation because DOE described these capabilities using TSPA input parameters and intermediate results, and analyses based on the TSPA models and parameters.

The time period over which the waste form and waste package internals limit the release of radionuclides is described in SAR Section 2.1.2.2.3, in which DOE described the degradation rates of the different waste forms. DOE supplemented this information in DOE Enclosure 5 (2009an), which identifies important processes controlling releases at different times in a discussion of selected TSPA realizations and described how DOE expects the significance of various processes (e.g., solubility, radionuclide sorption) to change with time. SAR

Section 2.1.2.2.6 described the change in the effectiveness of the capability by providing intermediate results from the TSPA that provided the time-dependent performance of the engineered barrier system to retain selected radionuclides. This information reasonably describes the time period over which the waste form and waste package internal components perform their barrier functions because the significance of different processes at different times is discussed.

DOE identified the sources of uncertainty in the barrier capabilities in SAR Section 2.1.2.2.4. These include, for example, uncertainties in the source term, in the evolution of in-package chemistry, in waste form degradation rates, and in radionuclide solubilities and sorption behaviors. In SAR Section 2.1.2.2.6, DOE described the effect of uncertainty on predictions of the ability of the waste form and waste package internals to limit the release of radionuclides from a failed waste package by showing uncertainty bounds on the amount of selected radionuclides retained within the engineered barrier system. These discussions show that DOE considered uncertainty in its descriptions of barrier capability because DOE identified sources of uncertainty and the effect of these uncertainties on radionuclide release.

DOE summarized the technical basis of the barrier capability description of the engineered barrier system, which included the waste form and waste package internal components, in SAR Section 2.1.3.2. DOE based its description of barrier capability of the waste form and waste package internal components on seven TSPA submodels: (i) the in-package water chemistry model described in SAR Section 2.3.7.5; (ii) three waste form degradation models described in SAR Sections 2.3.7.7, 2.3.7.8, and 2.3.7.9; (iii) the dissolved concentrations limits model described in SAR Section 2.3.7.10; (iv) the colloidal radionuclide availability model described in SAR Section 2.3.7.11; and (v) the engineered barrier system flow and transport model described in SAR Section 2.3.7.12.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.2 and 2.1.3.2 with SAR Sections 2.3.7.5 and 2.3.7.7 through 2.3.7.12 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.2 and 2.1.3.2 with the results of the seven waste package internals TSPA submodels DOE described in SAR Sections 2.3.7.5 and 2.3.7.7 through 2.3.7.12, and notes that the radionuclide release rate estimates are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Section 2.1.2.2 are consistent with the technical basis of the waste package internals TSPA submodels.

The NRC staff's evaluation of the technical bases for these models is documented in TER Section 2.2.1.3.4. The NRC staff notes in TER Section 2.2.1.3.4 that DOE provided technical bases for the waste package internals in TSPA submodels and for the ranges of values for the radionuclide release rates used in the performance assessment that address their intended use. The NRC staff therefore determines that DOE's technical bases for the description of the barrier capability of the waste form and waste package internals are commensurate with the barrier capability described in SAR Section 2.1.2.2 and in DOE Enclosures 1 and 5 (2009an).

In summary, the NRC staff notes that the capability of the waste form and waste package internal components to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis for the performance assessment abstraction of radionuclide release and engineered barrier system transport properties.

2.2.1.1.3.2.7 Engineered Barrier System: Emplacement Pallet and Invert

DOE discussed the capabilities of the emplacement pallet and invert in SAR Section 2.1.2.2 and in DOE Enclosure 1 (2009an). DOE did not consider the emplacement pallet and invert to be important to waste isolation, and DOE, therefore, did not provide a detailed description of their capabilities.

DOE identified a potential barrier capability of the emplacement pallet to reduce diffusive releases from the engineered barrier system by preventing contact between the waste package and invert, thereby reducing diffusive releases, but explained that this capability was not included in the TSPA. The NRC staff notes that the mechanical integrity of the emplacement pallet affects the analyses of damage to waste packages during seismic events. Therefore, the NRC staff has separately evaluated DOE's assumptions about the potential mechanical stability of the emplacement pallet in TER Section 2.2.1.3.2.3.3.

DOE explained that the invert contributes to barrier capability because low diffusion rates and potential sorption of radionuclides in the crushed tuff ballast slow the release rate of radionuclides from the waste package to the unsaturated rock beneath the drift. However, DOE determined that the delaying effect in the invert is not significant over long time frames, so DOE classified the invert as not important to barrier capability. The NRC staff notes that the potential for precipitation of low solubility radionuclides, a process discussed under the waste package internal components, is also a potential capability that can be associated with the invert. However, this process is likely to be more effective within the failed waste package, where water flows are typically much lower, than within the more dilute conditions of the invert. The NRC staff's evaluation of the technical basis for the dissolved concentrations limits model, described in SAR Section 2.3.7.10, is documented in TER Section 2.2.1.3.4.

The NRC staff reviewed DOE's description of barrier capability for the emplacement pallet and invert. Based upon examination of the engineered barrier system radionuclide transport abstraction described in SAR Section 2.3.7 and evaluated in TER Section 2.2.1.3.4, the NRC staff notes that DOE appropriately considered the emplacement pallet and invert as not important to waste isolation, because the capabilities of the invert are not significant in comparison with other components of the engineered barrier system identified in SAR Table 2.1-1.

2.2.1.1.3.2.8 Lower Natural Barrier: Unsaturated Zone Below the Repository

The NRC staff reviewed DOE's description of the barrier capability of the unsaturated zone below the repository. DOE identified the unsaturated zone beneath the repository as important to waste isolation because it prevents or substantially reduces the rate of movement of radionuclides (SAR Table 2.1-1). DOE provided a qualitative description of the barrier capabilities of the unsaturated zone below the repository in SAR Section 2.1.2.3.1. In SAR Section 2.1.2.3.6 and in DOE Enclosures 1, 6, and 7 (2009an), DOE quantified the barrier capability with calculations of radionuclide travel times and reduction of radionuclide activity between the repository and the water table. DOE provided information that describes a capability to substantially reduce the rate of movement of radionuclides from the repository to the water table.

DOE explained in SAR Section 2.1.2.3.1 that downward flow from the repository occurs primarily in well-connected fracture networks in the Topopah Spring welded tuff. DOE explained in SAR Section 2.1.2.3.6 that radionuclides leaving the emplacement drift invert will enter either the repository host rock matrix (primarily under nondripping conditions, where advective flows through the invert are negligible) or the fractures (primarily under dripping conditions, where advective flows through the invert are relatively high). In DOE's unsaturated zone transport abstraction (SAR Section 2.3.8), radionuclides may be retarded by sorption in the matrix but not in fractures. However, radionuclides can migrate from fractures to the rock matrix by matrix diffusion. DOE identified matrix diffusion, coupled with sorption in the matrix, as contributing to barrier performance in the fracture-dominated flow paths. DOE explained in SAR Section 2.1.2.3.1 and SAR Section 2.1.2.3.6 that radionuclide travel times through the lower unsaturated zone are fast in the northern part of the repository area because fracture-dominated flow from the repository host rock encounters a low-permeability, sparsely fractured rock unit, the zeolitic Calico Hills nonwelded tuff, and the flow is diverted laterally along the interface into transmissive faults that connect with the water table. In contrast, in the southern part of the repository area, the fracture-dominated flow from the repository host rock passes into the vitric Calico Hills nonwelded tuff, a permeable rock unit that is dominated by matrix flow conditions. Low flow velocities and the opportunity for sorption in the rock matrix result in long transport times through the unsaturated zone in the southern part of the repository area, particularly for radionuclides that can undergo sorption in the matrix.

DOE provided quantitative information on the barrier capability of the unsaturated zone below the repository in SAR Sections 2.1.2.3.6 and 2.3.8.5.4 and in DOE Enclosures 1, 6, and 7 (2009an). Using results from the TSPA model with median parameter values, SAR Figures 2.3.8-43 through 2.3.8-49 indicate that the barrier performance of the lower unsaturated zone varies according to the location of the radionuclide release (i.e., northern or southern part of the repository area) and the mode of release from the repository drift into the unsaturated zone (i.e., into fractures or matrix). DOE explained in SAR Sections 2.1.2.3.6 and 2.3.8.5.4 and in DOE Enclosures 1, 6, and 7 (2009an) that fracture flow dominates in the welded tuffs beneath the northern part and matrix flow dominates in the vitric Calico Hills tuff beneath the southern part of the repository. Radionuclides released from a northern location will therefore tend to reach the water table much faster than those released from a southern location. However, initial releases into the rock matrix will result in slow travel times regardless of release location. For example, DOE calculated that the median travel time of an unretarded tracer (Tc-99) through the lower unsaturated zone in the northern area is about 20 years for releases into fractures and about 5,000 years for releases into the matrix. For a southern release location, the calculated median travel times to the water table are slow regardless of whether releases are into fractures or the matrix, with either release mode resulting in a median arrival time of about 2,000 years (SAR Figure 2.3.8-49). Analyses documented in DOE Enclosures 1, 6, and 7 (2009an) showed that radioactive decay in the unsaturated zone coupled with a combination of matrix diffusion and sorption in the northern repository area and sorption in the vitric Calico Hills tuff layer in the southern repository area would substantially reduce releases of sorbing, short-lived radionuclides such as Cs-137. For longer lived radionuclides, DOE's analyses showed that sorption slows but does not prevent their transport through the unsaturated zone. For example, DOE Enclosure 7, Tables 1-5 through 1-8 (2009an) indicated that the unsaturated zone beneath the repository (northern and southern areas combined) reduced the release of long-lived radionuclides such as Np-237 (weakly sorbing) and Pu-242 (moderately to strongly sorbing) from the engineered barrier system to the saturated zone by about 30–50 percent during the 10,000-year period and by about 5 to 30 percent over a million-year time frame.

On the basis of its review of information DOE provided in SAR Section 2.1.2.3, the NRC staff notes that DOE's description of the barrier capability of the unsaturated zone below the repository is consistent with the results from the performance assessment calculation because

DOE described these capabilities using analyses based on TSPA models and input data as well as by reference to intermediate results from the performance assessment model.

In SAR Section 2.1.2.3.3 and DOE Enclosure 7, Section 1.3 (2009an), DOE discussed the time period over which the unsaturated zone functions as a barrier. DOE stated that the hydrogeology and physical characteristics of the lower natural barrier, which includes the unsaturated zone below the repository, are not expected to significantly change within 10,000 years after closure. DOE assumed that the intrinsic hydrologic, geologic, and geochemical characteristics of the lower natural barrier will not change significantly after 10,000 years following closure. DOE expects changes in the unsaturated zone capability to be associated with projected increases in percolation and in the water table elevation due to changes in climate. DOE Enclosure 7 (2009an) indicated that the relative barrier capability of the unsaturated zone decreases compared to the saturated zone for the post-10,000-year period because of faster travel times in the unsaturated zone. Information in DOE Enclosures 6 and 7 (2009an) stated that the barrier capability of the unsaturated zone is more pronounced for the initial 10,000-year time frame than for a million-year time frame because sorption slows but does not prevent the release of long-lived sorbing radionuclides to the saturated zone. DOE stated that delay times on the order of 1,000 years significantly affect short-lived radionuclides and that even the transported mass of long-lived radionuclides may be diminished by long travel times in the unsaturated zone. This information reasonably describes the time period over which the unsaturated zone performs its stated barrier functions because it identifies which aspects of the capability will change and which will remain constant.

DOE discussed and evaluated the uncertainties in the unsaturated zone in SAR Sections 2.1.2.3.4 and 2.3.8.5.5. DOE attributed the main uncertainties for barrier capability to (i) the variability of site characteristics and future climate and (ii) applicability of the models and assumptions used to estimate the performance of the repository system (SAR Section 2.1.2.3.4). Some examples of the uncertain characteristics included percolation flux, the extent of fracture–matrix interaction, matrix diffusion coefficients, and radionuclide distribution coefficients. DOE incorporated uncertainty in the TSPA unsaturated zone transport model by using sampled probabilistic distributions for parameter uncertainty and by using assumptions in models that would not overestimate performance. DOE presented the impact of various uncertainties in SAR Figures 2.3.8-50 to 2.3.8-62. It supplemented this information with discussions of sensitivity analyses for various parameters in DOE Enclosure 7, Section 1.2 (2009an). DOE considered uncertainty in its descriptions of barrier capability for the unsaturated zone below the repository because DOE described specific sources of uncertainty and indicated the performance impact of different sources of uncertainty.

In SAR Section 2.1.3.3, DOE summarized the technical basis of the barrier capability description of the lower natural barrier, which includes the unsaturated zone below the repository. DOE based its description of barrier capability of the unsaturated zone below the repository on the unsaturated zone flow model described in SAR Section 2.2.2.4 and on the unsaturated zone transport model described in SAR Section 2.3.8.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.3.1 and 2.1.3.3 with SAR Section 2.3.8 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.3.6 and 2.1.3.3 with the results of the unsaturated zone transport model described in SAR Section 2.3.8 and notes that the estimates of radionuclide travel time and reduction in radionuclide activity within the unsaturated zone are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Sections 2.1.2.3.1 and 2.1.2.3.6 are consistent with the technical bases of the unsaturated zone transport model.

The NRC staff documents its evaluation of the technical basis for the unsaturated zone flow model abstraction in TER Section 2.2.1.3.6 and for the unsaturated zone transport model abstraction in TER Section 2.2.1.3.7. The NRC staff notes in TER Section 2.2.1.3.7 that DOE provided reasonable technical basis for the unsaturated zone transport. Therefore, DOE's technical bases for the description of the barrier capability of the unsaturated zone below the repository are commensurate with the barrier capability described in SAR Section 2.1.2.3.1, SAR Section 2.1.2.3.6, and in DOE Enclosures 1, 6, and 7 (2009an).

In summary, the NRC staff notes that the capability of the unsaturated zone below the repository to prevent or substantially reduce the rate of movement of radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis for the performance assessment.

2.2.1.1.3.2.9 Lower Natural Barrier: Saturated Zone

The NRC staff reviewed the description of the barrier capability of the saturated zone. DOE described the barrier capabilities of the saturated zone qualitatively in SAR Section 2.1.2.3.2 and used median transport times and reduction of radionuclide activity between the water table below the repository footprint and the accessible environment to quantify the barrier capability in SAR Section 2.1.2.3.6 and in DOE Enclosures 1, 6, and 7 (2009an). The NRC staff determined that DOE identified a capability of the barrier to substantially reduce the rate of movement of radionuclides from the water table below the repository footprint to the accessible environment.

In SAR Section 2.1.2.3.2, DOE explained that the saturated zone component of the lower natural barrier flows initially through approximately 12–14 km [7.4–8.7 mi] of fractured volcanic rocks. Beyond this distance, flow is predominantly within a saturated layer of alluvium. DOE explained that the flow in the fractured volcanic aquifers occurs primarily in the fractures. DOE explained that hydraulic conductivities are much lower in the matrix of the volcanic tuffs than in the fractures, because the rock matrix is more porous than the fractures. These relative properties support exchange of radionuclides between the fractures and matrix through matrix diffusion. Hence, diffusion into the matrix followed by matrix sorption function to delay radionuclide transport to the accessible environment. DOE explained that flow and transport occur in the intergranular pores of the alluvial sediments after leaving the fractured volcanic aquifer. Because of the low water velocity, the rate of radionuclide movement is slow, allowing more time for sorption to occur onto the mineral surfaces to further delay radionuclide transport to the accessible environment. DOE also explained that the presence of colloids also affects the rate of movement of radionuclides in the saturated zone. Radionuclides embedded in or irreversibly sorbed onto colloids are retarded when the associated colloids are temporarily filtered from transport. Radionuclides that are sorbed reversibly to colloids are delayed by matrix diffusion in the volcanic aquifers and by sorption in the alluvial sediments.

DOE provided quantitative information on the barrier capability of the saturated zone in SAR Sections 2.1.2.3.6 and 2.3.9.3.4.1. SAR Figures 2.3.9-16 and 2.3.9-45 through 2.3.9-47 illustrated the combined effects of matrix diffusion and sorption in delaying radionuclide transport to the accessible environment. Median transport times ranged from about 10 to 10,000 years for nonsorbing radionuclides (e.g., Tc-99) and from 100 to 100,000 years for moderately sorbing radionuclides (e.g., Np-237). Median transport times generally exceeded

10,000 years for highly sorbing radionuclides (e.g., Pu-239). The median transport times for radionuclides irreversibly attached onto colloids ranged from 100 to 600,000 years. In DOE Enclosures 6 and 7 (2009an), DOE used TSPA results to provide quantitative information on the barrier capability of the saturated zone in terms of reduction of radionuclide activity between the release from the unsaturated zone into the water table and the release into the accessible environment. DOE presented information on the performance of the saturated zone in Tables 1-5 through 1-8. This information indicates that DOE expects activities of soluble, short half-life radionuclides (e.g., Cs-137 and Sr-90) to drop by 100 percent during transport to the accessible environment. For radionuclides with moderate to strong sorption and long half-life (e.g., Np-237 and Pu-242), DOE calculated the activities to drop by 70 to 98 percent during the 10,000-year period and by 20 to 50 percent during the post-10,000-year period over the transport time in the saturated zone to the accessible environment.

On the basis of its review of the DOE description of the barrier capabilities of the saturated zone in SAR Section 2.1.2.3, the NRC staff notes that DOE's description of the barrier capability for the saturated zone is consistent with the results from the performance assessment calculation because DOE describes the capabilities using TSPA intermediate results, supported by analyses based on TSPA models and parameters.

DOE provided information on the time period over which the saturated zone performs its intended function in SAR Section 2.1.2.3.3 and DOE Enclosure 7, Section 1.3 (2009an). Additional information on the time period over which the saturated zone functions as a barrier is contained in SAR Section 2.3.9.3.4.1. DOE stated that the hydrogeology and physical characteristics of the lower natural barrier, which includes the saturated zone, are not expected to change in any significant way within 10,000 years after closure. DOE assumed that the intrinsic hydrologic, geologic, and geochemical characteristics of the lower natural barrier will not change significantly after 10,000 years following closure. DOE addressed changes in the barrier function of the saturated zone by reference to an expected increase in groundwater recharge under projected wetter future climate conditions, resulting in a rise in the water table and increased groundwater flow. DOE did not expect these changes in groundwater flow to change the processes of sorption and matrix diffusion that control radionuclide transport to the accessible environment. DOE explained that sorption increases the barrier capability because it delays the release and allows for radioactive decay within the natural system to reduce the radionuclide mass in the system. Information in DOE Enclosures 6 and 7 (2009an) showed that the barrier capability of the saturated zone is more pronounced for the initial 10,000-year time frame than for a million-year time frame. DOE provided information that shows sorption slows but does not prevent the release of long-lived sorbing radionuclides to the saturated zone and that the effects of a delay are more pronounced in the initial 10,000 years after closure. DOE showed that delay times on the order of 1,000 years significantly affect short-lived radionuclides and that even the transported mass of long-lived radionuclides may be diminished by long travel times in the saturated zone. The NRC staff notes that this information reasonably describes the time period over which the saturated zone performs its stated barrier functions because it identifies which aspects of the capability will change and which will remain constant.

In SAR Section 2.1.2.3.4, DOE described the uncertainty in the barrier capability in terms of the conceptual and numerical models, observational data, and parameters used to represent water flow and radionuclide transport processes in the saturated zone. Some examples of the uncertain characteristics include groundwater-specific discharge, porosity, the spatial variation of aquifer properties, matrix diffusion coefficients, and radionuclide distribution coefficients. DOE incorporated parameter uncertainty in the TSPA saturated zone transport model through various probabilistic distributions. Effects of transport parameter uncertainty on radionuclide breakthrough at the accessible environment are presented in SAR Section 2.3.9.3.4.1 and SAR Figures 2.3.9-16 and 2.3.9-45 through 2.3.9-47. DOE also provided quantitative evaluations of the impacts of barrier uncertainties on saturated zone flow processes in SAR Section 2.3.9.2.3.4. DOE supplemented this information with discussions of sensitivity analyses for various parameters in DOE Enclosure 7, Section 1.2 (2009an). The NRC staff notes DOE has considered uncertainty associated with the modeling of water flow and radionuclide transport processes in its descriptions of barrier capability because DOE described specific sources of uncertainty and indicated the performance impact of different sources of uncertainty.

In SAR Section 2.1.3.3, DOE summarized the technical basis of the barrier capability description of the lower natural barrier, which includes the saturated zone. DOE based its description of barrier capability of the saturated zone on the saturated zone flow model described in SAR Section 2.3.9.2 and the saturated zone transport model described in SAR Section 2.3.9.3.

The NRC staff compared the technical basis descriptions in SAR Sections 2.1.2.3.2 and 2.1.3.3 with SAR Section 2.3.9 and notes that the technical basis descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative barrier capability description for the saturated zone in SAR Sections 2.1.2.3.6 and 2.1.3.3 with the results of the saturated zone transport model described in SAR Section 2.3.9 and notes that the estimates of radionuclide travel time and reduction in radionuclide activity within the saturated zone are consistently represented among these SAR sections. Therefore, the technical bases for the description of barrier capability in SAR Sections 2.1.2.3.2 and 2.1.2.3.6 are consistent with the technical basis of the saturated zone flow and transport models.

The NRC staff's evaluation of the technical bases for these models is documented in TER Sections 2.2.1.3.8 and 2.2.1.3.9, respectively. The NRC staff notes in TER Sections 2.2.1.3.8 and 2.2.1.3.9 that DOE provided technical bases for the saturated zone flow and transport models that address their intended use. Therefore, DOE's technical bases for the description of the barrier capability of the saturated zone are commensurate with the barrier capability described in SAR Section 2.1.2.3.2, SAR Section 2.1.2.3.6, and in DOE Enclosures 1, 6, and 7 (2009an).

In summary, the NRC staff notes that the capability of the saturated zone to prevent or substantially reduce the rate of movement of radionuclides from the Yucca Mountain repository to the accessible environment is reasonably described and that the technical basis for the barrier capability is based on and consistent with the technical basis for the performance assessment.

2.2.1.1.4 NRC Staff Conclusions

2.2.1.1.4.1 Identification of Barriers

As discussed in TER Section 2.2.1.1.3.1, DOE has identified specific features and components that are relied upon for repository performance. DOE has linked these features and components to a description of their capability. These features and components include at least one from the engineered system and one from the natural system. Therefore, the NRC staff notes that DOE has identified the barriers, consistent with applicable guidance in the YMRP.

2.2.1.1.4.2 Description of Barrier Capability to Isolate Waste

Upper Natural Barrier

The NRC staff reviewed the description of the barrier capabilities of the upper natural barrier. This barrier comprises two features: (i) the topography and surface soils at the repository location and (ii) the unsaturated zone above the repository. On the basis of the evaluations documented in TER Sections 2.2.1.1.3.2.1 and 2.2.1.1.3.2.2, NRC staff notes that the capability of the upper natural barrier has been reasonably described. The descriptions address a capability to prevent or substantially reduce the rate of movement of water from the Yucca Mountain repository to the accessible environment. The descriptions of the capabilities are consistent with the results from the Total System Performance Assessment in that they refer by reference to component-specific intermediate results from the performance assessment or by analyses based on models and data used in the performance assessment. Information on the time period over which this upper natural barrier feature performs its intended function has been provided. DOE has reasonably described the uncertainty associated with the barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, DOE has reasonably described the capability of the upper natural barrier to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

Engineered Barrier

The NRC staff reviewed the description of the barrier capabilities of the engineered barrier system. This barrier comprises five components: (i) the emplacement drift, (ii) the drip shield, (iii) the waste package, (iv) the waste form and waste package internals, and (v) the emplacement pallet and invert. On the basis of the evaluation documented in TER Sections 2.2.1.1.3.2.3 through 2.2.1.1.3.2.7, the NRC staff notes that the capability of the engineered barrier system has been reasonably described. The descriptions identify a capability to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment. The described capabilities are consistent with the results from the Total System Performance Assessment in that they are described by reference to component-specific intermediate results from the performance assessment or by analyses based on models and data used by the performance assessment. Information on the time period over which the features of the engineered barrier system perform their intended functions has been provided. DOE has reasonably addressed uncertainty in the description of the barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, DOE has reasonably described the capability of the engineered barrier system to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

Lower Natural Barrier

The NRC staff reviewed the description of the barrier capabilities of the lower natural barrier. This barrier comprises two features: the unsaturated zone below the repository and the saturated zone. On the basis of the evaluation documented in TER Sections 2.2.1.1.3.2.8 and 2.2.1.1.3.2.9, the NRC staff notes that the capability of the lower natural barrier has been reasonably described. The described capabilities address a capability to prevent or substantially reduce the rate of movement of radionuclides from the Yucca Mountain repository to the accessible environment. The described capabilities are consistent with the results from the Total System Performance Assessment in that they are described by reference to input data or component-specific intermediate results from the performance assessment or by analyses based on models and data used in the performance assessment. Information on the time period over which the features of the lower natural barrier system perform their intended functions has been reasonably addressed. DOE has reasonably addressed uncertainty in the description of the barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, DOE has reasonably described the capability of the lower natural barrier to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

2.2.1.1.4.3 Technical Basis for Barrier Capability

The SAR presents an overview of the technical bases for the models used to represent the performance of the barriers in the TSPA. This overview, summarized in SAR Section 2.1 and more fully documented in SAR Section 2.3, identifies the types of field investigations, laboratory studies, analog studies, literature surveys, and other technical approaches used to develop the conceptual TSPA model components. The NRC staff notes that the technical bases for the descriptions of barrier capability summarized in TER Sections 2.2.1.1.3.2.1 through 2.2.1.1.3.2.9 are consistent with the technical bases of the abstraction models described in SAR Section 2.3 and evaluated in TER Sections 2.2.1.3.1 through 2.2.1.3.9. The technical basis is commensurate with the barrier capability described in SAR Sections 2.1 and 2.3 and in DOE responses to NRC requests for additional information documented in DOE (2009an).

2.2.1.1.4.4 NRC Staff Conclusions

The NRC staff notes that the multiple barriers of the repository, as described in DOE's SAR and other information DOE submitted, are consistent with the guidance in the YMRP. NRC staff also notes that the repository includes both natural and engineered barriers and that the technical basis for each barrier's capability, as discussed in this chapter, is reasonable.

2.2.1.1.5 References

CNWRA and NRC. 2008aa. "Risk Insights Derived From Analyses of Model Updates in the Total-system Performance Assessment Version 5.1 Code." ML082240343. San Antonio, Texas: CNWRA.

DOE. 2009an. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1." Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bo. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (June 1) J.R. Williams to J.H. Sulima (NRC). ML091530403. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bu. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1." Letter (April 16) J.R. Williams to J.H. Sulima (NRC). ML091070088. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cb. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (June 5) J.R. Williams to J.H. Sulima (NRC). ML091590581. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

SNL. 2008ad. "Postclosure Nuclear Safety Design Bases." ANL–WIS–MD–000024. Rev. 01. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

(Intentionally Left Blank)

CHAPTER 2

2.2.1.2.1 Scenario Analysis

2.2.1.2.1.1 Introduction

This chapter provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE) scenario analysis used in its performance assessment. The NRC staff evaluated information in the Safety Analysis Report (SAR) (DOE, 2009av) as supplemented by the DOE responses to the staff's requests for additional information (RAIs) (DOE, 2009ab,ae,af,ah–aj,al,bo,bv–bz,ca–cj,co,cq,gp,gq, 2010ad,ah).

A performance assessment is a systematic analysis that answers the following triplet risk questions: What can happen? How likely is it to happen? What are the resulting consequences? Scenario analysis answers the first question: What can happen? A scenario is a well-defined, connected sequence of features, events, and processes (FEPs) that can be interpreted as an outline of a possible future condition of the repository system. Thus, a scenario analysis identifies possible ways in which a geologic repository environment can evolve so that a defensible representation of the system can be developed to estimate consequences. The goal of scenario analysis is to ensure that no important aspect of the potential high-level waste repository is overlooked in the evaluation of its safety.

A scenario analysis is generally composed of four parts (Nuclear Energy Agency, 2001aa). First, a scenario analysis identifies FEPs relevant to the geologic repository system. Second, in a process known as screening, the scenario analysis evaluates and identifies FEPs for exclusion from or inclusion into the performance assessment calculations. Third, included FEPs are considered to form scenarios and scenario classes (i.e., related scenarios) from a reduced set of events. Fourth, the scenario classes are screened for implementation into the performance assessment.

Consistent with this general approach and the review areas in the Yucca Mountain Review Plan (YMRP) Section 2.2.1.2.1 (NRC, 2003aa), the NRC staff evaluates DOE's scenario analysis in four separate sections [Technical Evaluation Report (TER) Sections 2.2.1.2.1.3.1 to 2.2.1.2.1.3.4]. TER Section 2.2.1.2.1.3.1 evaluates both DOE's methodology to develop a list of FEPs and its list of FEPs. In TER Section 2.2.1.2.1.3.2, the NRC staff evaluates DOE's screening of its list of FEPs, including DOE's technical bases for the exclusion of FEPs. DOE's formation of scenario classes and the exclusion of classes in DOE's performance assessments are evaluated in TER Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4, respectively.

The NRC staff's evaluation of DOE's methodology and conclusions on the probability of events included in the performance assessments is presented in TER Section 2.2.1.2.2. That section is aimed at the second risk triplet question: How likely is it to happen? The NRC staff's evaluation of DOE's model abstraction is documented in TER Sections 2.2.1.3.1–2.2.1.3.14 and Sections 2.2.1.4.1–2.2.1.4.3. These sections focus on the included FEPs and the third risk triplet question (What are the resulting consequences?) and present the NRC staff's evaluation of the adequacy of assessment of consequences of included FEPs and scenario classes used in DOE's performance assessments.

2.2.1.2.1.2 Evaluation Criteria

A performance assessment is an analysis that identifies the features, events, processes (except human intrusion), and sequences of events and processes (except human intrusion) that might affect the Yucca Mountain disposal system and their probabilities of occurring. A functional overview of the performance assessment used to demonstrate compliance with the postclosure performance objectives is presented in 10 CFR 63.102(j). 10 CFR 63.102(j) also contains criteria for including FEPs [those expected to materially affect compliance with 10 CFR 63.113(b) or be potentially adverse to performance] in the performance assessment. 10 CFR 63.102(j) provides that events (event classes or scenario classes) that are very unlikely (less than 1 chance in 10,000 over 10,000 years) can be excluded from the analysis.

The postclosure performance objectives of 10 CFR 63.113 stipulate that a performance assessment must be used to demonstrate compliance with (i) the individual protection standard after permanent closure (10 CFR 63.311); (ii) the human intrusion standard (10 CFR 63.321 and 63.322); and (iii) the separate standards for protection of groundwater (10 CFR 63.331). Criteria for a performance assessment used to calculate annual doses for 10,000 years are presented in 10 CFR 63.114(a). 10 CFR 63.114(a)(4) states that the performance assessment consider only features, events, and processes consistent with the limits on performance assessment specified at 10 CFR 63.342. 10 CFR 63.114(a)(5)–(6) defines criteria for inclusion of the features, events, and processes into the performance assessment (specific features, events, and processes need detailed evaluation if the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, for 10,000 years after disposal, would be significantly changed by their omission). 10 CFR 63.114(b) calls for performance assessments consistent with paragraph (a) of that section to be considered sufficient for the performance assessment for the period of time after 10,000 years and through the period of geologic stability.

According to 10 CFR 63.342(a), the performance assessment for 10,000 years after disposal need not include features, events, and processes with less than 1 chance in 100 million per year of occurring. Also, 10 CFR 63.342(a) provides that the performance assessments need not evaluate the impacts resulting from any feature, event, and process or sequence of events and processes with a higher chance of occurring if the results of the performance assessments would not be changed significantly in the initial 10,000-year period after disposal. Thus, 10 CFR 63.342 defines the conditions for exclusion of FEPs on the basis of probability or consequence.

For performance assessments for human intrusion and groundwater protection, 10 CFR 63.342(b) provides that unlikely features, events, and processes or sequences of events and processes (i.e., those that are estimated to have less than 1 chance in 100,000 per year of occurring and at least 1 chance in 100 million per year of occurring) can be excluded from the performance assessment.

10 CFR 63.342(c) specifies how to project the continued effects of FEPs beyond 10,000 years in the performance assessment models. 10 CFR 63.342(c) requires a performance assessment to project the continued effects of the features, events, and processes included in 10 CFR 63.342(a) beyond the 10,000-year postdisposal period through the period of geologic stability. 10 CFR 63.342(c)(1) requires that the effects of seismic and igneous activity scenarios are evaluated subject to the probability limits in 10 CFR 63.342(a) for very unlikely features, events, and processes, or sequences of events and processes.

10 CFR 63.342(c)(1)(i) states that the seismic analysis may be limited to the effects caused by damage to the drifts in the repository, failure of the waste packages, and changes in the elevation of the water table under Yucca Mountain (i.e., the magnitude of the water table rise under Yucca Mountain).

10 CFR 63.342(c)(1)(ii) specifies limitations for the igneous activity analysis and igneous event. 10 CFR 63.342(c)(2) requires an assessment of the effects of climate change and specifies that the climate change analysis may be limited to the effects of increased water flow through the repository as a result of climate change, and the resulting transport and release of radionuclides to the accessible environment. In addition, 10 CFR 63.324(c)(2) specifies that the nature and degree of climate change may be represented by constant-in-time climate conditions. 10 CFR 63.342(c)(3) specifies an assessment of the effects of general corrosion on engineered barriers and allows for either the use of a constant representative corrosion rate throughout the period of geologic stability or a distribution of corrosion rates correlated to other repository parameters.

Guidance in YMRP Section 2.2.1.2.1.3, p. 2.2-9 provides that specific FEPs and scenario classes can be excluded on the basis that they are specifically ruled out by regulation or are contrary to stated regulatory assumptions. For example, 10 CFR 63.305 defines characteristics of the reference biosphere. Thus FEPs that are contrary to definitions in 10 CFR 63.305 can be excluded.

2.2.1.2.1.3 Technical Evaluation

DOE summarized in SAR Section 2.2.1 its five-step scenario analysis method used to develop a performance assessment model: (i) identification and classification of a list of FEPs, (ii) evaluation of the FEPs for inclusion or exclusion from the performance assessment model, (iii) formation of scenario classes, (iv) screening of scenario classes, and (v) definition of the implementation of scenario classes in the performance assessment model and documentation of the treatment of included FEPs. The first four steps are evaluated in this section. Step five is evaluated in TER Sections 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3.

The staff's evaluation of the scenario analysis follows the methodologies and acceptance criteria identified in YMRP Section 2.2.1.2.1, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The guidance in YMRP Section 2.2.1.2.1 provides four criteria for a scenario analysis.

- The identification of a list of features, events, and processes is adequate.
- Screening of the list of features, events, and processes is appropriate.
- Formation of scenario classes using the reduced set of events is adequate.
- Screening of scenario classes is appropriate.

Additionally, YMRP Section 2.2.1 provides guidance to the NRC staff on how to apply risk information in its review of the DOE analyses. Following the YMRP guidance, the NRC staff considered DOE's risk information (derived from DOE's treatment of multiple barriers) and risk insights in SAR Section 2.4.2.2.1.2. The level of detail of the NRC staff's review of particular parts of the scenario analysis is based on (i) the risk insights DOE developed; (ii) consideration of the risk insights identified in NRC Appendix D (2005aa), as updated (CNWRA and NRC, 2008aa); and (iii) detailed process modeling efforts, laboratory and field experiments, and natural analog studies.

Because the purpose of the first acceptance criterion in YMRP Section 2.2.1.2.1 is to ensure the completeness and comprehensiveness of the FEPs list, the NRC staff did not consider risk information in its review of DOE's identification of a list of FEPs (TER Section 2.2.1.2.1.3.1). In TER Section 2.2.1.2.1.3.2, the NRC staff reviewed DOE's screening of the list of FEPs following the second acceptance criterion in YMRP Section 2.2.1.2.1 and as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The acceptance criterion includes the following three separate aspects: (i) all features, events, and processes that are excluded are identified; (ii) justification for each excluded feature, event, and process is provided [an appropriate justification for excluding FEPs is that either the feature, event, and process is specifically excluded by regulation; probability of the feature, event, and process (generally an event) falls below 10^{-4} in 10,000 years; or omission of the FEP does not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment]; and (iii) an adequate technical basis for each excluded feature, event, and process is provided. The NRC staff evaluated the technical bases of the 222 excluded features, events, and processes. In reviewing the technical basis for exclusion of FEPs, the NRC staff focused in greater detail on items that were deemed to have the largest impact on risk and used progressively less detail on items that were considered to have lower to negligible impact on risk. For example, drift collapse is a process that could affect multiple aspects of the repository (e.g., temperature, moisture distribution, rock loads acting on the drip shield, response of the drip shield subjected to seismic excitations) (Ofoegbu, et al., 2007aa) and that could affect the performance of multiple engineered barrier components that impact risk (NRC, 2005aa). Accordingly, the NRC staff devoted greater effort to evaluate the technical basis for exclusion of the Drift Collapse FEP. On the other hand, a number of FEPs were deemed to be not risk significant (e.g., Meteorite Impact, Copper Corrosion in the Engineered Barrier System), and these FEPs were evaluated in less detail. In this chapter, in general, the FEPs discussed are those for which the NRC staff issued RAIs to DOE to supplement the technical bases for exclusion.

2.2.1.2.1.3.1 Identification of a List of Feature, Events, and Processes

Identification of a list of FEPs is the initial step in the scenario analysis and is aimed at assembling a list that includes all FEPs with the potential to influence repository performance. This technical evaluation of the identification of the list of FEPs follows the methodology established in YMRP Section 2.2.1.2.1.2, p. 2.2-7.

DOE summarized the process to identify the list of FEPs in SAR Section 2.2.1.1.1 and in SNL (2008ac). DOE has published two major versions of the list of FEPs for the Yucca Mountain project: the FEPs list for site recommendation and the SAR FEPs list. DOE stated that the site recommendation FEPs list was developed based on a Nuclear Energy Agency compilation of FEPs, supplemented with Yucca Mountain project literature, information in analysis reports, technical workshops, and reviews and resulted in a collection of 328 FEPs considered in the site recommendation Total System Performance Assessment, as outlined in SNL p. 6-1 (2008ac).

DOE stated that the site recommendation FEPs list was further refined to enhance classification strategies and to achieve a consistent level of detail among FEPs, and that additional FEPs were identified on the basis of audits and technical information updates subsequent to the site recommendation, such as changes in design parameters. DOE stated that to verify comprehensiveness in the list of FEPs, an alternative list was developed using a top-down functional analysis of the repository (SNL, 2008ac). Each function was divided into smaller, more specialized functions until a level of detail was attained comparable to the existing list of FEPs. This alternative list was then compared to the SAR FEPs list to build confidence that the SAR FEPs list was indeed complete or to identify missing FEPs. DOE further compared the SAR FEPs list to a version of an international list of FEPs by the Nuclear Energy Agency, Organisation for Economic Co-operation and Development (OECD), Appendix D (Nuclear Energy Agency, 2000aa) to inquire about the completeness of the list of FEPs. DOE noted that the International FEPs Database was updated in 2006 (Nuclear Energy Agency, 2006aa); however, DOE concluded that the update did not present additional scope beyond the FEPs already addressed in the SAR FEPs list (SAR p. 2.2-8) and SNL Appendix F (2008ab).

DOE stated that further analyses were applied to address changes in the regulations and in the design of the repository and disposal packages. The final count of FEPs is 374. DOE stated that the iterative approach, including expanding on the existing FEPs list, brainstorming, multiple reviews by subject matter experts, top-down elicitation from an independent classification scheme, and use of the Yucca Mountain project analyses support the conclusion that the SAR FEPs list was complete, as described in SNL p. 6-4 (2008ac).

The NRC staff evaluated the adequacy of the list of FEPs. The FEPs were classified by technical area (Leslie, 2010aa), following a similar approach as in NRC Table 5.1.2.1-2 (2005aa). In numerous instances, the same FEP was classified as pertaining to several technical areas, to cover broad aspects, consequences, and couplings associated with that FEP (Leslie, 2010aa). The objective of assigning an FEP to multiple technical areas was to attain a thorough and integral review of the list of FEPs covering multiple technical perspectives and to facilitate indentifying aspects potentially overlooked by the existing FEPs. The NRC staff evaluated the description of the scope for the individual FEPs, the screening decision of the individual FEPs, the technical basis for excluding FEPs, and the disposition for the included FEPs. The NRC staff's review of the identification of the list of FEPs was based on knowledge gained reviewing the Yucca Mountain site and regional characterization data, including previous independent Yucca Mountain-related studies, and DOE's description of the modes of degradation, deterioration, and alteration of the engineered barriers. A previous NRC staff's review of DOE's identification of FEPs (Pickett and Leslie, 1999aa) was also considered. The NRC staff also used available, internationally developed generic lists of FEPs (Nuclear Energy Agency, 1997ab) to evaluate the completeness of the DOE list of features, events, and processes.

The NRC staff noted in NRC Section 5.1.2.1.4.1 (2005aa) that the features, events, and processes list for site recommendation was based on a Nuclear Energy Agency international database of FEPs (Nuclear Energy Agency, 1997ab). Using SNL Appendix G (2008ab), a cross comparison of the FEPs lists for site recommendation, and the SAR, the NRC staff noted that the SAR FEPs list encompasses the FEPs list for site recommendation. Using SNL Appendix F (2008ab), tables that map the SAR FEPs into the Organisation for Economic Co-operation and Development FEPs, and vice versa, the NRC staff confirmed that the SAR list of FEPs encompasses the Organisation for Economic Co-operation and Development features, events, and processes.

Summary of NRC Staff's Review for Identification of a List of Feature, Events, and Processes

Because the SAR FEPs list encompasses generic comprehensive lists of internationally approved FEPs (Nuclear Energy Agency, 2006aa, 2000aa, 1997ab) and is consistent with the site characterization data and the SAR design features, DOE's list of FEPs is reasonable.

DOE's complete listing of FEPs considered (SAR Table 2.2-5) includes FEPs that address potentially disruptive events related to igneous activity (e.g., FEP 1.2.04.03.0A and FEP 1.2.04.07.0A); seismic shaking (e.g., FEP 1.2.03.02.0A and FEP 1.2.03.02.0B); tectonic evolution (e.g., FEP 1.2.01.01.0A); climatic change (e.g., FEP 1.3.01.00.0A and FEP 1.3.07.02.0B); and criticality (e.g., FEP 2.1.14.16.0A and FEP 2.1.14.17.0A).

On the basis of the information in SAR Section 2.2.1.1.1 and references cited therein, the NRC staff notes that

- SAR Table 2.2-5 contains a complete list of features, events, and processes related to the geologic setting or the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) that have the potential to influence repository performance

- The list of features, events, and processes in SAR Table 2.2-5 is consistent with the site characterization data

- The feature, event, and process list includes, but is not limited to, potentially disruptive events related to igneous activity, seismic shaking, tectonic evolution, climatic change, and criticality

Therefore, DOE's identification of a list of features, events, and processes is reasonable.

2.2.1.2.1.3.2 Screening of the List of Features, Events, and Processes

Screening of the list of FEPs is aimed at identifying FEPs that should be evaluated in detail in the performance assessment due to their clear potential to influence repository performance. The technical evaluation of the screening of the list of FEPs follows the methodology established in YMRP Section 2.2.1.2.1.2, p. 2.2-7, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab).

DOE summarized the screening of FEPs in SAR Section 2.2.1.2. SAR Table 2.2-5 summarized the screening decision (to include or exclude) for each FEP and the justification for exclusion. SAR Table 2.2-5 cited other SAR tables summarizing the technical basis for including the FEPs and also cited SNL (2008ab) as the document that detailed the technical basis for excluding FEPs.

In SAR Section 2.1.2.2, DOE described that the regulations call for inclusion of certain FEPs in the performance assessment evaluations that are conducted to compute doses for the period after 10,000 years after disposal, but within the period of geologic stability. DOE described that FEPs associated with these criteria were evaluated for inclusion in the appropriate performance assessments. DOE stated that no changes to screening decisions were necessary to address the inclusion of FEPs specified by the proposed 10 CFR 63.342(c)(1), (2), and (3). The DOE restated this issue in two parts. First, DOE stated that the FEPs specified by regulation to be included in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability, are also included in the performance assessments for the 10,000 years after disposal. Second, DOE stated that FEPs that are excluded from the performance assessments for the 10,000 years after disposal remain excluded in the performance assessments for the period after the first 10,000 years after disposal, but within the period of geologic stability. In SAR

Section 2.1.2.2 DOE identified the specific included FEPs that are consistent with the proposed 10 CFR 63.342(c)(1)(i) regulatory criteria and described that excluded FEP 1.2.03.02.0B, Seismic Induced Rockfall Damages Engineered Barrier System (EBS) Components, was also evaluated with respect to proposed 10 CFR 63.342(c)(1)(i). DOE Enclosure 6 (2009cb) identified that the excluded FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, is consistent with proposed 10 CFR 63.342(c)(1)(i). In SAR Section 2.1.2.2 DOE also identified the included FEPs that are consistent with proposed 10 CFR 63.342(c)(1)(ii), 10 CFR 63.342(c)(2), and 10 CFR 63.342(c)(3).

DOE Enclosure 6 (2009cb) performed a detailed comparison between the proposed 10 CFR Part 63 (NRC, 2005af) rule and the final 10 CFR Part 63 rule that became effective on April 13, 2009, and identified material changes in the final rule and how those changes may materially impact information in the SAR and supporting documents. DOE Enclosure 6 (2009cb) specifically discussed (i) arithmetic mean of projected doses; (ii) water table rise due to seismic activity; (iii) changes to the range of deep percolation rates; and (v) dosimetry. In addition, DOE Enclosure 6, Section 1.6 (2009cb) evaluated the potential impacts of all changes identified in DOE Enclosure 6, Appendix, Table A-1 (2009cb) and concluded that none of the conclusions in the SAR require modification as a result of the final rule.

With respect to procedural safety controls and design configuration controls, DOE stated that SAR Table 2.2-3 identified FEPs that relate to parameters requiring procedural safety controls or design configuration control to ensure that the performance assessment analysis basis is met. SAR Table 1.9-9 summarized the parameters requiring such controls. DOE noted that the repository design (as defined in the included FEP 1.1.07.00.0A, Repository Design, and the controlled design parameters in SAR Table 2.2-3) was used to define the initial state or boundary conditions in the models and the analyses that are abstracted in the postclosure performance assessment. DOE also stated in SAR Section 2.2.1.2 that controlled parameters and the repository design were used as a basis for describing other FEPs and as a basis for screening decisions of included and excluded FEPs. According to DOE, SAR Table 1.9-9 presented design control parameters that describe the bases for the repository design.

The NRC staff initially noted that a number of FEPs lacked adequate technical basis to support the DOE's exclusion conclusion. DOE supplemented (DOE, 2009ab,ae,af,ah–aj,al,bv–bz, ca–ci,co,cq,gp,gq, 2010ad,ah) the information in SNL (2008ab) to respond to the NRC staff's requests for additional information. The NRC staff reviewed the information in SNL (2008ab), the supporting analyses referenced therein, and the DOE responses to the requests for additional information (DOE, 2009ab,ae,af,ah–aj,al,bo,bv–bz,ca–cj,co,cq,gp,gq, 2010ad,ah).

NRC Staff's Evaluation

The NRC staff used the YMRP, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab), to evaluate whether the screening of the list of FEPs is appropriate. The YMRP provides guidance on (i) whether DOE identified all FEPs that have been excluded, (ii) whether DOE provided justification for exclusion of those FEPs, and (iii) whether DOE provided adequate technical basis for exclusion of those FEPs.

The NRC staff noted that DOE has identified all FEPs related to either the geologic setting or to the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) that have been excluded. SAR Table 2.2-5 listed all of the FEPs DOE considered, and it identified the excluded FEPs. With regard to point (ii) , the NRC staff determines that DOE's criteria for exclusion on the basis of low probability, low consequence, or by regulation is reasonable because these criteria are consistent with criteria for scenario analysis discussed in TER Section 2.2.1.2.1.2. DOE clearly stated in SAR Table 2.2-5 the criterion it applied for exclusion of each FEP; thus, DOE has provided reasonable justification for the excluded FEPs (on the basis of low probability, low consequence, or by regulation).

With regard to point (iii), the NRC staff's evaluation of the technical basis for the exclusion of FEPs is provided in the remainder of this section. First, the NRC staff evaluates DOE's information on screening of FEPs for the period after 10,000 years after disposal, but within the period of geologic stability. Then, the NRC staff evaluates DOE's information on screening of FEPs for the 10,000 years after disposal.

The NRC staff reviewed DOE's information in SAR Section 2.1.2.2 on screening of FEPs for the period after 10,000 years after disposal, but within the period of geologic stability using the guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The NRC staff notes the following DOE statements are reasonable: (i) FEPs included in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability, are also included in the performance assessments for the 10,000 years after disposal and (ii) FEPs that are excluded from the performance assessments for the 10,000 years after disposal remain excluded in the performance assessments for the period after the first 10,000 years after disposal, but within the period of geologic stability. The NRC staff evaluates the adequacy of DOE's analyses of included FEPs in TER Sections 2.2.1.3.1–2.2.1.3.14 and Sections 2.2.1.4.1–2.2.1.4.3. The NRC staff evaluates DOE's technical basis for the exclusion of FEPs in this section.

The NRC staff evaluated all of the descriptions, screening decisions, and screening justifications (i.e., DOE's technical basis) of the FEPs DOE classified as excluded (Leslie, 2010aa; DOE excluded a total of 222 FEPs). Only those FEPs that the NRC staff identified as requiring additional information or clarification are specifically discussed in this section, and as described next, DOE's clarifications were sufficient for the NRC staff to complete its evaluation. The discussed FEPs correspond to approximately 10 percent of the total number of excluded FEPs and are summarized later in this section under individual FEP headings (with the exception of the criticality FEPs that are all reviewed under the criticality FEPs heading). For each FEP discussed under an individual FEP heading, and for the group of criticality FEPs, the NRC staff's evaluation includes a summary of DOE's information followed by the NRC staff's review of the technical basis for the exclusion of the individual FEP (or FEPs, in the case of the criticality FEPs). Additional subheadings, where needed to enhance the readability, are used to identify the NRC staff's review or to identify the review of specific technical aspects associated with the individual FEP (or FEPs, in the case of the criticality FEPs).

On the basis of the NRC staff's review (Leslie, 2010aa), the NRC staff notes that the excluded FEPs that are not discussed in this section (i.e., the remaining 90 percent of the excluded FEPs) were adequately defined and that adequate technical bases were provided to support DOE's exclusion decision. The following are examples of FEPs that are not discussed in this section and that were excluded by regulation, probability, and consequence.

Exclusion by Regulation

For each FEP that DOE excluded by regulation and that is not explicitly addressed in TER Section 2.2.1.2.1.3.2, the NRC staff checked to see whether DOE provided an appropriate regulatory citation to exclude the FEP. The NRC staff reviewed the adequacy of the technical basis (SNL, 2008ab) for each FEP DOE excluded by regulation to ensure the technical basis was consistent with the cited regulation. The NRC staff notes that DOE provided adequate technical bases for these FEPs.

Exclusion by Probability

For each FEP that was excluded by probability and that is not explicitly addressed in TER Section 2.2.1.2.1.3.2, the NRC staff reviewed the adequacy of technical basis (SNL, 2008ab) provided for each FEP. DOE's technical basis for excluding low probability FEPs by showing that the annual probability is less than 10^{-8} in the first 10,000 years is reasonable. For example, for FEP 1.5.01.01.0A, Meteorite Impact, DOE provided a quantitative analysis of meteorite impact probability and used crater information, repository site information, and a design parameter to demonstrate that the low probability criterion would be met. DOE's basis for FEP 1.5.01.01.0A is reasonable because DOE used impact rates that are consistent with data from available literature. DOE overestimated the impact footprint of the repository, and DOE's analysis was consistent with repository site characteristics and the repository's design. Similarly, for the other FEPs excluded on the basis of low probability, the NRC staff notes that DOE provided reasonable technical bases.

Exclusion by Low Consequence

For each FEP that was excluded by low consequence and that is not explicitly addressed in TER Section 2.2.1.2.1.3.2, the NRC staff reviewed the adequacy of technical basis (SNL, 2008ab) provided for each FEP. For those FEPs DOE excluded on the basis of low consequence, DOE provided suitable technical basis by showing that omission of the FEP does not significantly change the magnitude and time of the resulting radiological exposures to reasonably maximally exposed individual, or radionuclide releases to the accessible environment. For example, DOE excluded FEP 1.1.02.00.0A, Chemical Effects of Excavation and Construction in the Engineered Barrier System, on the basis that (i) relevant construction materials are design-controlled parameters or subject to controls and (ii) analyses show negligible impact from engineered materials on the groundwater chemistry (SNL, 2008ab). DOE's technical basis for excluding FEP 1.1.02.00.0A is reasonable because DOE described the analyses which evaluated the effects and identified the controls that would be imposed (e.g., constraints will be imposed on the administrative control of tracers, fluids, and materials; construction materials; and committed materials). Similarly, for the other FEPs excluded on the basis of low consequence, the NRC staff notes that DOE provided reasonable technical bases.

The NRC staff notes that DOE's use of repository design and controlled parameters to define FEPs is reasonable; this use is also reasonable as a technical basis for screening. The NRC staff notes DOE's proposed approach to control parameters identified in SAR Table 1.9-9, through use of management systems is an adequate basis for the repository design considered in the development of screening justifications for FEPs. The NRC staff notes that it is appropriate to use the repository design to define the scope of FEPs as well as to define initial states or boundary conditions of systems analyzed in the performance assessment. In addition, SAR Table 2.2-3 is an adequate mechanism to track interdependencies and identify FEPs with screening technical bases that would need reevaluation if some parameters were to depart from initial design considerations. The NRC staff notes that the design information and the design assumptions are appropriate.

FEP 1.1.01.01.0B, Influx Through Holes Drilled in Drift Wall or Crown

DOE excluded Influx Through Holes Drilled in Drift Wall or Crown on the basis of low consequence SNL (2008ab) and supplemented its technical basis for exclusion in DOE Enclosures 2 and 7 (2009cb). As defined by DOE, FEP 1.1.01.01.0B addresses the potential of openings (or holes) that may be drilled through the drift walls or crown to promote flow or seepage into the drifts and onto the waste packages. These holes may be drilled for a variety of reasons including, but not limited to, rock bolt and ground support, monitoring and testing, or construction-related activities. For boreholes, FEPs 1.1.01.01.0B and 2.1.06.04.0A, Flow Through Rock Reinforcement Materials in the Engineered Barrier System, according to DOE's definitions, cover similar processes and features because open space will be present in boreholes regardless of whether rock bolts degrade.

DOE stated in SNL (2008ab) that boreholes into the walls of emplacement drifts will be drilled for ungrouted rock bolts and ground support and identified in SAR Table 2.2-3 that Control Parameters 01-15 and 01-16 apply to FEPs 1.1.01.01.0B and 2.1.06.04.0A. Using a modified version of the seepage model used for the performance assessment in BSC Sections 6.5 and 6.6.4 (2004be), DOE examined the potential for liquid water flowing into open rock bolt boreholes that extend vertically upwards from the drift crown. DOE concluded, supported by numerical simulations, that boreholes have only a minor effect on seepage, increasing the predicted seepage rates by less than 2 percent compared to seepage simulations without rock bolts. DOE based this result on the following considerations and assumptions: (i) an open borehole without grout acts as a capillary barrier to unsaturated flow; (ii) the cross-sectional area of the rock bolt borehole, onto which flow may be incident, is small; and (iii) water that may have seeped into the borehole can imbibe back into the rock matrix elsewhere along the borehole length. On the basis of this analysis, DOE concluded that the presence of boreholes drilled in the drift wall or crown would not have a significant effect on seepage into drifts, and excluded the FEP Influx Through Holes Drilled in Drift Wall or Crown from the performance assessment model on the basis of low consequence.

NRC Staff's Review

The NRC staff assessed the seepage modeling evaluation for boreholes and considered observations from ambient and thermally perturbed field tests. Given the widespread presence of boreholes in the drifts, the NRC staff performed a more detailed evaluation of the exclusion basis for FEP 1.1.01.01.0B. The NRC staff estimated that there will be approximately 26 rock bolts per waste package in the circumferential extent of the drift wall used to estimate seepage. This number was derived from the repository design whereby rock bolts will be installed with circumferential and axial spacing of 1.25 m [4.1 ft] in a 240° arc around the drift periphery and above the invert structure (SAR Section 1.3.4.4.1).

DOE's previously listed considerations and assumptions (i), (ii), and (iii) [(i) an open borehole without grout acts as a capillary barrier to unsaturated flow; (ii) the cross-sectional area of the rock bolt borehole, onto which flow may be incident, is small; and (iii) water that may have seeped into the borehole can imbibe back into the rock matrix elsewhere along the borehole length] are reasonable bases for supporting the exclusion of the feature, event, and process. The NRC staff evaluated the adequacy of the technical bases supporting the considerations and assumptions that DOE provided in a discussion of the results from the seepage modeling exercise. The NRC staff analyzed the consistency of observations from field tests and site characterization with results from the DOE seepage modeling exercise for boreholes. First, observations of temperature fluctuations from the heater tests may be indicative of water flowing in boreholes at host-rock temperatures above boiling (Green, et al., 2008aa). Second, posttest visual observations indicate water entered the drifts and thus did not absorb back into the wall of the borehole, though timing and temperature at which this occurred is not known (Green, et al., 2008aa). Third, secondary mineralization in large aperture (open) fractures that DOE attributed to percolating water under ambient conditions suggests capillary diversion may not keep water from entering boreholes. Fourth, observations of liquid water in the drift during the passive test may be explained and modeled as vapor flux through fractures from within the host rock and condensation in cooler rock spots (Salve and Kneafsey, 2005aa) rather than by a capillarity-based seepage model of liquid water dripping into drifts.

In response to an NRC staff's request for additional information, DOE Enclosures 2 and 7 (2009cb) supplemented the technical basis and provided additional information on the relationship of field observations to flow in boreholes and seepage into drifts. DOE framed the supplemental information in terms of effects during thermal and ambient periods and relied on a total-system performance perspective; in particular, on the drip shield seepage barrier function. For the thermal period, DOE Enclosure 7 (2009cb) pointed out the drip shield function of diverting water that has entered the drift. According to DOE, the drip shields are expected to divert water during the thermal period and are expected to fail by general corrosion and cease to be a barrier against seepage well after the thermal pulse has dissipated and the system has returned to ambient conditions. DOE Enclosure 5 (2009bo) referred to supplemental analyses showing that radionuclide releases are relatively insensitive to the occurrence of seepage in the event of seismic damage to waste packages under intact drip shields.

DOE also analyzed other cases where the drip shield may fail during the thermal period (e.g., early failure, seismic fault displacement, and seismic ground motion modeling cases) and concluded that in none of those cases would borehole effects on seepage significantly alter the dose estimates. For the early failure case, DOE referred to the low contribution of this case to the total mean dose and stated that changes in reflux would marginally affect the dose. For the fault displacement modeling case, DOE stated that full collapse of the drift is generally associated with fault displacement, and thus, thermal reflux in open boreholes has a negligible effect on the mean annual dose from seismic fault displacement, as described in DOE Enclosure 7 (2009cb). In the seismic modeling case, DOE described that the drip shield would fail only for large magnitude seismic events, which would be accompanied by large rockfall and borehole collapse. Therefore, thermal reflux in such boreholes would have a negligible effect on dose estimates, as detailed in DOE Enclosure 7 (2009cb). The exclusion of FEP 1.1.01.01.0B during the thermal period is reasonable because the drip shield protects the waste package against seepage and because of the relatively weak dependence of the mean dose on seepage in DOE's performance assessment.

The NRC staff's evaluation for the ambient period focused on the potential increase in water entering the drift, either by seepage from boreholes or by vapor flux through boreholes. For seepage from boreholes, FEP 1.1.01.01.0B (SNL, 2008ab) cited sensitivity analyses suggesting a 2 percent increase in seepage compared to domains without boreholes. This difference would readily fall in the range of uncertainty incorporated in the seepage results for the performance assessment. Furthermore, boreholes are not a factor in the seismic ground motion and igneous intrusion modeling cases, which are the two largest contributors to dose. According to the DOE model, seismic events would cause significant collapse of the host rock above drifts (e.g., SAR Figure 2.1-14; DOE, 2008ab) by the time drip shields are expected to fail by general corrosion (e.g., SAR Figure 2.1-11; DOE, 2008ab), thus eliminating any potential effect of boreholes on seepage. In addition, the DOE abstraction for the igneous intrusion modeling case eliminates the seepage barrier capability of drifts. For the vapor flux through boreholes, DOE Enclosure 2 (2009cb) described that the magnitude of the vapor flux asymptotically decreases from the latter stages of the thermal period to the ambient period. Consistent with DOE's technical basis is the suggestion that the water observed in the drift of the passive test would coincide with early entrance of vapor into the drifts. This flux will decrease with time as the entire system (drift and rock) moves closer to hydrological equilibrium. Using its condensation model, DOE stated that the magnitude of the condensation flux estimated for later times (after the thermal period) is much less than the estimated seepage flux derived from the seepage model. To provide confidence in the condensation flux estimate for early times, DOE stated that a conservative assumption of relative humidity at the drift wall of 100 percent was used in the condensation model. It is reasonable to use this assumption to estimate condensation on the basis of the condensation model review in TER Section 2.2.1.3.6.3.5, where the NRC staff noted that the condensation model was reasonable for its intended purpose within the context of the performance assessment model. DOE's technical basis for excluding FEP 1.1.01.01.0B from the performance assessment is reasonable on the basis of low consequence.

FEP 1.1.03.01.0A, Error in Waste Emplacement

DOE excluded Error in Waste Emplacement on the basis of low consequence (DOE, 2009av; SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 1 (2009af) and DOE Enclosure 2(2009cq). FEP 1.1.03.01.0A, according to DOE, refers to deviations from the design or errors in waste emplacement that could affect long-term performance of the repository. DOE identified two types of waste emplacement errors: the first concerns spacing of waste packages and the second concerns emplacement of a waste package on a fault. DOE described controls that will be carried out to restrict by detection, evaluation, and mitigation the probability of both types of waste emplacement errors. These controls include controlled parameters and management controls. DOE also assessed the potential consequences of undetected and unmitigated waste emplacement errors in DOE Enclosure 1 (2009af) and DOE Enclosure 2 (2009cq). DOE assessed the probability for waste package misplacement and violation of the thermal limits for the repository. DOE estimated the mean number of misplaced waste packages to be less than one. DOE compared the consequences of waste emplacement errors to the consequences of the waste package early failure modeling case and the seismic fault displacement modeling case; DOE included both of these in the performance assessment.

NRC Staff's Review

The NRC staff reviewed the screening justification and the technical basis for excluding FEP 1.1.03.01.0A. DOE provided an exclusion justification of low consequence in DOE Enclosure 1 (2009af) and DOE Enclosure 2 (2009cq). The NRC staff used its knowledge of the proposed repository operations and repository performance assessments to assess potential consequences. On the basis of DOE's description of the feature, event, and process and the NRC staff's knowledge, both types of waste emplacement errors that DOE identified are sufficient to evaluate potential consequences on repository performance from waste emplacement errors. The NRC staff assessed whether the controls DOE identified in DOE Enclosure 1, Tables 1 and 2 (2009af) and DOE Enclosure 2 (2009cq) were adequate to limit errors in emplacing the waste. The NRC staff notes DOE's proposed controls are appropriate. DOE's assessment of the probabilities of undetected and unmitigated waste emplacement errors is reasonable because the rates are consistent with error rates for comparable controlled activities reviewed in TER Section 2.2.1.2.2.3. DOE identified that both the probability and consequence of waste emplacement error is less than that assessed in the waste package early failure model case. Both the low probability and the comparison to the early failure case to assess waste emplacement spacing errors are reasonable because in the early failure case, DOE assumed damaged waste packages, while waste emplacement errors do not necessarily imply the presence of a damaged waste package leading to radionuclide release. On the basis of the low consequences associated with the seismic fault displacement modeling case (SAR Section 2.4.2.2.1.2.2.2) and the NRC staff's independent assessment of the risk from seismic fault displacement (Waiting, et al., 2003aa), DOE's conclusion that waste emplacement on a fault is of low consequence is reasonable. Therefore, the justification for exclusion is reasonable. The NRC staff notes it is reasonable to exclude FEP 1.1.03.01.0A, Error in Waste Emplacement, on the basis of low consequence.

FEP 1.2.04.07.0B, Ash Redistribution in Groundwater

DOE excluded Ash Redistribution in Groundwater on the basis of low consequence. According to DOE's definition of FEP 1.2.04.07.0B, during a volcanic eruption, magma may interact with waste packages, resulting in erupted deposits of volcanic ash contaminated with radionuclides. DOE limited FEP 1.2.04.07.0B to the leaching of radionuclides from the ash and their subsequent transport in groundwater through the subsurface to the reference boundary or location of the RMEI. DOE considered other processes, such as ash remobilization by wind, in separate FEPs.

The DOE volcanic eruption model considers the mass and types of waste impacted by erupted magma (a maximum of seven damaged waste packages), the fraction of waste-containing magma that is incorporated into a tephra plume, and the fraction of the tephra plume that is deposited near the eruptive vent (i.e., in or near the repository footprint) (SNL, 2008ag). In contrast, the DOE igneous intrusion model assumes that (i) all waste packages in the repository are compromised by an igneous intrusion and (ii) the subsequent release of waste is not reduced by the amount that could be transported to the surface in an accompanying eruption (SNL, 2007ab). DOE reasoned that because eruptive events are always associated with intrusive events, the potential dose consequences from radionuclides leached into groundwater from volcanic ash are small compared with the consequences of exposing the same inventory of radionuclides to seepage due to igneous intrusion (SNL, 2008ab). However, for the modeled fraction of volcanic ash that is deposited near the accessible environment boundary, the groundwater transport path is short compared to the transport path for radionuclides released from the repository. Short transport pathways to the accessible environment boundary have potential dose consequences for short-lived, high-activity radionuclides such as Cs-137, Sr-90, Am-241, and Pu-238, which have half-lives on the order of decades or hundreds of years. The short-lived radionuclides are important contributors to dose at early times in the volcanic eruption modeling case (e.g., SAR Figure 2.4-32).

In DOE Enclosure 1 (2009ab), DOE supplemented its technical basis for exclusion and assessed the relative importance of leaching and groundwater transport of contaminated ash deposited near the accessible environment. DOE's supporting calculation addressed differences in travel times depending on where the ash was deposited within the drainage basin of Fortymile Wash (i.e., very short flow paths for leaching of ash deposited near the accessible environment boundary and longer flow paths for ash deposited upstream). The transport calculation included the effects of radioactive decay and retardation of radionuclides and indicated that leaching of contaminated ash would not contribute significantly to mean annual dose compared to the volcanic eruption modeling case, as detailed in DOE Enclosure 1, Figure 1 (2009ab).

NRC Staff's Review

On the basis of the NRC staff's comparison of the conditions and assumptions in DOE's volcanic eruption modeling case and igneous intrusion modeling case, the NRC staff notes, for volcanic ash deposited near the eruptive vent, that DOE's technical basis for exclusion of FEP 1.2.04.07.0B in SNL (2008ab) is reasonable. Potential dose consequences near the eruptive vent are bounded by the dose consequences of igneous intrusion because the groundwater transport pathways are similar for both examples. For leaching and transport from ash deposited in Fortymile Wash, as described in DOE Enclosure 1 (2009ab), the NRC staff's evaluation focused on short-lived radionuclides because of their high radioactivity levels that could dominate dose estimates for this fast pathway scenario. The NRC staff contrasted DOE's computations for this fast pathway scenario with the contribution to dose from the same short-lived radionuclides in DOE's volcanic eruption modeling case (e.g., SAR Figure 2.4-32). The NRC staff noted that DOE provided the technical basis for excluding FEP 1.2.04.07.0B with respect to leaching and transport in short groundwater flow pathways because the supporting calculations showed that the transport of the short-lived radionuclides was delayed sufficiently, even in the relatively short groundwater transport pathways in Fortymile Wash, to allow radioactive decay to significantly diminish their potential contribution to dose. The models DOE used for those supporting calculations are consistent with those models that the NRC staff reviewed for water flow paths and radionuclide transport in TER Sections 2.2.1.3.8 and 2.2.1.3.9, which the NRC staff notes are adequate in the context of DOE's performance assessments. Therefore, the model used in the supporting calculations to evaluate potential fast pathways is reasonable. DOE's technical basis for excluding FEP 1.2.04.07.0B on the basis of low consequence is reasonable.

FEP 1.2.07.01.0A, Erosion/Denudation

DOE excluded FEP 1.2.07.01.0A, Erosion/Denudation, from the performance assessment on the basis of low consequence. Erosion involves the transport of surficial material away from the site by mechanisms including glacial, fluvial, eolian, and chemical processes. As part of FEP 1.2.07.01.0A, DOE also considered processes such as weathering, mass wastage processes (e.g., landslides), and local uplift (SNL, 2008ab).

DOE cited site characterization studies concluding erosion ranging from 0.4 to 2.7 cm [0.16 to 1.06 in] in 10,000 years for bedrock outcrops and 0.2 to 6 cm [0.08 to 2.4 in] in 10,000 years for unconsolidated material in hillslopes. DOE concluded that the maximum expected erosion of 6 cm [2.4 in] in 10,000 years is consistent with existing surface irregularities and that erosion would be negligible compared with the minimum distance of 200 m [656.2 ft] from the ground surface to the repository emplacement areas (SNL, 2008ab). DOE considered the effect of erosion on the extent of net infiltration and concluded it was negligible. Further, DOE described that the homogenizing action of the Paintbrush nonwelded hydrogeologic unit would buffer any localized change in net infiltration (SNL, 2008ab). DOE stated that bedrock weathering could increase the soil thickness and decrease the net infiltration. Thus, disregarding weathering is a conservative approach. DOE referred to site characterization studies to conclude that processes such as landslides and debris flows do not play a significant role in the erosional regime at Yucca Mountain.

DOE stated that climatic conditions strongly influence erosional patterns, with deposition occurring during wetter periods and erosion occurring during drier periods. Because the 10,000-year period after closure is dominated by the glacial-transition climate (8,000 years of wetter climate), deposition is expected to be the dominant geomorphic process for the 10,000-year period after closure. DOE stated that deposition leads to soil buildup, and thus, disregarding deposition is conservative. Another process affecting erosion is uplift, and DOE stated that local rates of uplift are low, on the order of 0.01 mm/yr [3.94×10^{-4} in/yr].

NRC Staff's Review

On the basis of the analysis DOE developed, the technical basis for excluding FEP 1.2.07.01.0A, Erosion/Denudation, is reasonable. The erosion rates DOE quoted are consistent with site description data at BSC (2004bi) and are expected to cause negligible amounts of erosion in 10,000 years. DOE's conclusion that neglecting the effects of erosion in the performance assessment would not significantly affect the timing or magnitude of radionuclide releases into the accessible environment is reasonable. Therefore, the exclusion of FEP 1.2.07.01.0A, Erosion/Denudation, on the basis of low consequence is reasonable.

FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity

DOE excluded Hydrologic Response to Seismic Activity on the basis of low consequence (SNL, 2008ab), and the technical basis was supplemented as described in DOE Enclosure 19, (2009ab), DOE Enclosures 1–6 (2009by), DOE Enclosure 1 (DOE, 2009bz), DOE Enclosures 1–2 (2009ca), and DOE Enclosure 6 (2009cb). In supplementing the technical basis for this feature, event, and process, DOE also evaluated water table rise due to seismic activity beyond the 10,000-year postdisposal period through the period of geologic stability and included information on potential permanent changes in hydrologic properties in DOE Enclosure 6 (2009cb). According to DOE's definition of FEP 1.2.10.01.0A, seismic activity associated with fault movement may enhance existing or create new flow pathways or connections and barriers between stratigraphic units, or it may change the stress (and therefore fluid pressure) within the rock. These responses have the potential to change groundwater flow directions, water level, water chemistry, and temperature. Seismically induced changes to the local stress fields can cause a transient change in the water table elevations and lead to seismic pumping—a phenomenon DOE defined as the temporary change in water table elevation resulting from fault movement and the opening and closing of fractures during an earthquake.

The low consequence screening decision is based on DOE's conclusion that seismic events will result in relatively minor changes to the Yucca Mountain hydrologic system—changes that have no impact on repository performance. DOE's rationale is based on implicit assumptions of how the repository will respond to seismic loads typical for relatively large-magnitude western U.S. earthquakes, observational evidence from recent earthquakes, and modeling results used to support the National Research Council study (National Research Council, 1992aa) on the effects of earthquakes on the water table at Yucca Mountain.

In SNL (2008ab), DOE cited modeling investigations that have been conducted to estimate the hydrologic response (i.e., change in water table elevations), given predicted fault displacements in National Research Council Chapter 5 (1992aa). As described in SNL (2008ab) the National Research Council study estimated, using two fault displacement modeling approaches (i.e., a dislocation approach and a "changes in the regional stress" approach), that the maximum seismically induced water table rise over a 10,000-year period would be 17 m [56 ft] for the dislocation approach and 50 m [160 ft] for the regional stress approach. In addition, SNL (2008ab) described that the hydrologic effects of three seismic events in 1992 that were observed in groundwater monitoring wells at Yucca Mountain provide estimates of water-level fluctuations occurring in response to earthquakes. DOE examined the effects of the Landers–Big Bear–Little Skull Mountain earthquake sequence that occurred

June 28–29, 1992, and indicated that the water table rise observed at several Yucca Mountain vicinity monitoring wells ranged from 0.2 to 0.9 m [0.7 to 3 ft]. On the basis of the earthquake-caused water table change data and analyses in the National Research Council Study (1992aa), DOE concluded the maximum change will be no more than a 50-m [160-ft] water table rise beneath the repository.

SNL (2008ab) also cited Gauthier, et al., pp. 163–164 (1996aa), who analyzed the potential effects of seismic activity resulting from three fault displacement types (normal, listric, and strike-slip) with 1-m [0.3-ft] displacement and 30-km [19-mi] rupture length. Gauthier, et al. (1996aa) concluded that a strike-slip seismic event would cause a water table rise of 50 m [160 ft] within 1 hour and would return to steady-state conditions within 6 months. Other types and magnitudes of displacement were shown to cause smaller water table rises with similar transient durations.

DOE revised the rationale in SNL (2008ab) for excluding FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, in supplemental documents DOE Enclosure 19 (2009ab), DOE Enclosures 1–6 (2009by), DOE Enclosure 1 (DOE, 2009bz), DOE Enclosures 1–2 (2009ca), and DOE Enclosure 6 (2009cb). First, DOE drew a distinction between two modeling types it used to evaluate water table rise from seismic activity in the Yucca Mountain area: the regional stress change model and the dislocation model. In making this distinction, DOE emphasized the bounding nature of the regional stress change model; this model gave high values of predicted water table rise (higher than the dislocation model) that should be regarded as representative of the upper limits (bounds) of potential water table rise. DOE attributed these high estimates of seismically induced water table rise to a series of simplifying assumptions in the model. Using data from several studies since 1992, DOE cited evidence to suggest that the dislocation model more realistically represents the actual magnitude of seismically induced water table rise. DOE concluded that the water table rise values of the regional stress change model are overestimates of the seismically induced water table rise at Yucca Mountain.

Using the (bounding) regional stress change model and results from the Probabilistic Seismic Hazard Assessment (PSHA), DOE performed calculations to evaluate the potential of local (to Yucca Mountain) faults as sources for future water table rise at Yucca Mountain. Using the likely seismic characteristics of faults as given in the PSHA, DOE generated scenarios to calculate the values of maximum water table rise for each fault. Of 3,150 calculated scenarios, 13 generated water table rise exceeding 175 m [574 ft]. DOE then calculated the probabilities that such events will occur using the PSHA hazard probabilities. Although some of the probabilities are greater than the 10^{-8} per year threshold, DOE contended that because the regional stress change model overestimates water table rise, these results support excluding this feature, event, and process. Through the use of Probabilistic Fault Displacement Hazard Assessment results, DOE estimated that slip events with a 10^{-8} per year probability of exceedance would produce water table rise values between 30 and 122 m [100 and 400 ft].

DOE described that water table rises of these magnitudes are not sufficient to reach the proposed repository, even in the case of future wetter climate conditions. DOE estimated that the highest water table elevation beneath the repository footprint due to future wetter climate conditions would be limited to 850 m [2,790 ft] above sea level. This assumed water table elevation is generally consistent with results of a separate analysis by DOE that used the saturated zone site-scale flow model. This separate analysis evaluated the potential effects of a future wetter climate on saturated zone flow and estimated future climate-induced water table elevations as high as 875 m [2,870 ft] above sea level (SNL, 2007ax) beneath northwestern portions of the repository. Given that the range of repository drift elevations falls between 1,040 and 1,100 m [3,400 and 3,610 ft] above sea level, DOE concluded that water table depths under a future wetter climate would range between 187 and 250 m [620 and 820 ft] below the repository floor. Therefore, the additional transient water table rise due to a seismic event would remain below the repository drifts.

SNL (2008ab) addressed, as part of the FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, long-term changes in water table elevations that could be associated with seismic-induced permanent changes in regional permeability. SNL (2008ab) described that longer term changes are not expected to result from such permanent changes in stress, because the existing data do not show any relationship between the long-term state of stress and water table elevation. DOE Enclosure 6 (2009cb) described that the effects of seismic activity that could lead to permanent changes in hydrologic properties were evaluated (SNL, 2008ab) in the screening justifications for excluded FEPs 2.2.06.01.0A (Seismic Activity Changes Porosity and Permeability of Rock), 2.2.06.02.0A (Seismic Activity Changes Porosity and Permeability of Faults), and 2.2.06.02.0B (Seismic Activity Changes Porosity and Permeability of Fractures). These three FEPs were defined to address localized changes in porosity and permeability in intact rock, faults, and fractures and were excluded based on results from the drift-scale test, the PSHA, modeling and sensitivity studies, and information from the National Research Council (1992aa).

NRC Staff's Review

The NRC staff reviewed the information DOE provided in DOE Enclosure 19, (2009ab), DOE Enclosures 1–6 (2009by), DOE Enclosure 1 (DOE, 2009bz), DOE Enclosures 1–2 (2009ca), DOE Enclosure 6 (2009cb), and SNL (2008ab, 2007ax) and notes that exclusion of FEP 1.2.10.01.0A is supported by information and analyses in the SAR and supplemental documents, and that the technical basis for exclusion is appropriate for the following reasons.

First, DOE supported the poroelastic model and transient nature of any water-level changes due to an earthquake with observations from historical earthquakes, including earthquakes in the western United States and earthquakes in the vicinity of Yucca Mountain. DOE's conclusion that there are no permanent changes in water table elevations that could be associated with seismic-induced permanent changes in regional permeability is reasonable because existing data do not show any relationship between the long-term state of stress and water table elevation and because DOE Enclosure 6 (2009cb) identified that based on National Research Council (1992aa), earthquake-induced water table rise is expected to be transient. Thus, DOE's conclusion that any potential changes to the water table from earthquakes in the vicinity of Yucca Mountain are transient is reasonable. DOE's conclusion is reasonable that even in the least likely case of an earthquake that causes water levels to rise sufficiently to wet the waste packages, water levels would return to ambient elevations quickly, within a few years after the earthquake. In addition, the NRC staff also notes that the risk (probability-weighted consequences) would be negligible, because the likelihood of earthquakes with magnitudes large enough to induce changes in the water table is small (less than about 10^{-6}/yr). Thus, the potential impacts on repository performance would be negligible.

Second, the statement in DOE's supplemental assessments DOE Enclosure 19, (2009ab), DOE Enclosures 1 and 5 (2009by), DOE Enclosure 1 (DOE, 2009bz), and DOE Enclosures 1–2 (2009ca) that the analyses used to estimate seismically induced water table rise overestimate the extent of seismically induced water table rise is reasonable. The modeling and analyses the National Research Council Study (1992aa) and Kemeny and Cook (1992aa) relied on are based on assumed confined aquifer conditions. The water table below Yucca Mountain in the tuff aquifer is indicative of unconfined aquifer conditions. As DOE documented in DOE Enclosure 1 (2009ca), recent observations of changes to water table elevations in unconfined aquifers from large earthquakes in Taiwan and Japan were substantially smaller than the changes in the hydraulic head of nearby confined aquifers. DOE attributed differences in the reaction between confined and unconfined aquifers to the substantially smaller storability of confined aquifers.

Third, both the National Research Council (1992aa) and the Kemeny and Cook (1992aa) analyses relied on a regional stress change model. DOE's view is reasonable that of the two modeling approaches (i.e., the dislocation model and the regional stress change model), the regional stress change model overestimates the seismically induced water table rise. The NRC staff also notes that DOE's view that two simplifying assumptions in the regional stress change model—uniform stress changes throughout the rock body and uniform changes in pore pressures—cause this model to overestimate the seismically induced water table rise is reasonable. Because the regional stress change model's overestimates of seismically induced water table rise indicate the water table will remain below the level of waste emplacement drifts after an earthquake, even during future wetter climates, the NRC staff notes that DOE's technical basis supports the exclusion of FEP 1.2.10.01.0A from the performance assessment model.

Further, DOE's screening rationale is also applicable to the period of geologic stability, because DOE considered, in general, seismic events with recurrence rates of at least 10^{-8}/yr, as described in DOE Enclosure 6 (2009cb). The NRC staff notes that although DOE did not consider the information in National Research Council p. 94 (1995aa) (i.e., "Results indicate a probable maximum transient rise on the order of 20 m of less.") in determining the magnitude of water table rise from seismic activity for the beyond the 10,000-year postdisposal period through the period of geologic stability, DOE Enclosure 6 (2009cb) did consider results in the National Research Council (1992aa) investigations in determining the magnitude of the water table rise from seismic activity. Because the analyses used to estimate seismically induced water table rise overestimate the extent of seismically induced water table rise, including those identified in National Research Council (1992aa), and because DOE's technical basis supports the exclusion of FEP 1.2.10.01.0A from the performance assessment model for the initial 10,000-year period after repository closure, DOE's technical basis to exclude FEP 1.2.10.01.0A from the performance assessment analysis beyond the 10,000-year postdisposal period through the period of geologic stability is reasonable.

FEP 1.4.01.00.0A, Human Influences on Climate

DOE excluded Human Influences on Climate, due to future human activities, to avoid speculative prediction of changes to human behavior. DOE stated that present and past human influences on climate (which are within the scope of the included FEP 1.3.01.00.0A, Climate Change) are implicitly included in estimates of modern climate used in the performance assessment (SNL, 2008ab) and, as such, are evaluated in TER Section 2.2.1.3.5.

The NRC staff notes that the DOE approach to constrain the scope of the FEP to future changes in human activities is reasonable, as well as exclusion of the FEP from the performance assessment because speculative prediction of human behavior is avoided.

FEP 1.4.01.02.0A, Greenhouse Gas Effects

DOE excluded Greenhouse Gas Effects on climate, due to future human activities, to avoid speculative prediction of changes to human behavior. DOE constrained the scope of the feature, event, and process to future changes in human activities that may influence the concentrations of atmospheric gases. Present and past increases in greenhouse gases attributed to human activity are within the scope of the included FEP 1.3.01.00.0A, Climate Change. Those present and past changes are implicitly included in estimates of modern climate used in the performance assessment (SNL, 2008ab) and, as such, are evaluated in TER Section 2.2.1.3.5. .

The NRC staff notes that the DOE approach to constrain the scope of the FEP to future changes in human activities is reasonable, as well as exclusion of the FEP from the performance assessment because speculative prediction of human behavior is avoided.

FEP 1.4.07.03.0A, Recycling of Accumulated Radionuclides From Soils to Groundwater

DOE excluded Recycling of Accumulated Radionuclides from Soil to Groundwater on the basis of low consequence using a recycling model that estimated effects on the total system performance results (SNL, 2008ab). DOE supplemented its technical basis in DOE Enclosures 2–4 (2009af). DOE used this feature, event, and process to refer to the downward migration of contaminated irrigation water to the water table and the subsequent recapture and reuse (i.e., recycling) by irrigation wells within the contaminant plume that can potentially increase the concentration of radionuclides in the groundwater and dose to the reasonably maximally exposed individual. According to DOE, this contaminant concentration through recycling can occur only when the infiltrating irrigation water is applied within the capture zone of a pumping well that is also capturing all or part of the contaminant plume.

The DOE screening analysis for radionuclide recycling in groundwater is based on a model that assumes a single hypothetical water supply well with an uninterrupted withdrawal rate of 3.715×10^9 L [3,000 acre-ft] per year from the center of a contaminant plume. Capture zone dimensions for this hypothetical well are computed based on the local-groundwater-specific discharge and saturated aquifer thicknesses upgradient and downgradient of the well. DOE considered three mechanisms by which radionuclides can be lost from the recycling process: (i) irrigation water usage on fields located outside of the capture zone, (ii) residential water usage at locations outside of the capture zone, and (iii) erosion of soil from irrigated fields to locations outside of the recycling system. On the basis of current water usage in Amargosa Valley, about 90 percent of withdrawn water is used for irrigation. DOE's screening analysis concludes that recycling could increase the total mean annual dose by approximately 7 to 11 percent for the seismic ground motion and igneous intrusion scenarios for the 1-million-year simulation period and by an average of 11 percent for the 10,000-year simulation period (SNL, 2008ab), which is not significant compared with the range of uncertainty simulated by the Total System Performance Assessment model. On the basis of this result, DOE excluded this feature, event, and process from the performance assessment.

NRC Staff's Review

The NRC staff reviewed DOE's screening analysis [SNL, (2008ab); DOE Enclosures 2–4 (2009af)] and consulted available literature relevant to irrigation practices, infiltration of irrigation water, and methods for determining capture zone geometry. The NRC staff evaluated the reasonableness of DOE's supplemental information in DOE Enclosures 2–4 (2009af) that addressed the technical basis of the three aspects of DOE's screening analysis: (i) assumed capture zone geometry, (ii) assumed distances between the hypothetical pumping well and irrigated fields, and (iii) the assumption that radionuclides which reach the water table and are within the well capture zone are returned to the well volume without accounting for transport within the saturated zone does not underestimate doses at later times.

The assumed geometry of the capture zone for the hypothetical pumping well in DOE's analysis (SNL, 2008ab) was based on an idealized system of a pumping well applied to a background of uniform, parallel groundwater flow lines, whereas the observed pattern of water levels in the Amargosa region indicates a converging flow field. A converging flow field can lead to a wider capture zone compared to the one used in DOE's analysis in SNL (2008ab), which in turn could result in increased recycling and concentrations of radionuclides in groundwater. DOE showed, as identified in DOE Enclosure 2 (2009af), that the results of its screening analysis are not affected significantly when a converging flow field is considered. DOE's conclusion is reasonable on the basis that DOE showed that converging flow fields did not significantly change the capture fraction (i.e., the fraction of irrigation recharge that is captured by the reasonably maximally exposed individual's well).

The result of the screening model is strongly dependent on the capture fraction, which DOE calculated to be approximately 10 percent. This value is a reflection of the spatial distribution of irrigated fields relative to the steady-state capture zone (which DOE assumed to be located anywhere within the community). DOE used a probabilistic distribution based on evidence of field locations in the Amargosa Valley community and considered a single hypothetical water supply well. This approach tended to spread the distances between the fields and the well, potentially resulting in a relatively small capture fraction. Farmers might minimize the distance between the fields and the well as a cost-cutting approach. For example, in a study by Stonestrom, et al. (2003aa) on estimates of deep percolation, each of the three fields investigated had its own well for irrigation. A reduction in the distances of fields to the hypothetical pumping well could result in a greater well recapture fraction and increased radionuclide recycling. DOE explained in DOE Enclosure 3 (2009af) that the distances between irrigated fields and the well were not intended to represent actual distances. Rather, the screening analysis was a stylized approach constrained by requiring the pumping well to be at the location of highest concentration in the plume. DOE's supplementary analysis in DOE Enclosure 3 (2009af) was based on a model in which the pumping wells within and adjacent to the plume are coincident with irrigated fields that vary in location and pumping duration during a 10,000-year simulation period. This supplemental analysis, as identified in DOE Enclosure 4 (2009af), explicitly accounted for transport time of recycled irrigation water through the saturated zone before the water is potentially recaptured by other randomly located irrigation wells. The analysis indicated that the average increase in radionuclide concentrations due to recycling of pumped water was 4.9 percent for nonsorbing radionuclides and negligible for sorbing radionuclides. This updated model does not use the steady-state approach involving a single well intersecting the highest concentration of the plume as in the original model in SNL (2008ab). DOE concluded that the updated model is more reasonable and realistic, mimicking current practices.

To evaluate the case where the well intersects the highest concentration in the plume and irrigated fields are in proximity to the well, the NRC staff considered a hypothetical case where a well was used to irrigate a number of fields. The NRC staff considered the well located within the accessible environment and above the maximum plume concentration. If the irrigated fields were distributed in space at random, half of the fields would be located upstream from the well and half downstream. As an approximation, the NRC staff considered that upstream fields would be within the well capture zone and downstream fields outside the well capture zone. Therefore, in this simplified assessment, if pumping were to continue indefinitely with no soil erosion losses, a maximum of 50 percent of the radionuclides in the irrigated water could be recycled, causing concentrations of radionuclides to double at most. The NRC staff considers that a factor-of-two increase in the concentrations and dose consequences is a relatively moderate effect, given that this simplified analysis represents a hypothetical case and DOE's largest mean dose estimate is on the order of 0.01 msv [1 mrem] per year for the groundwater pathway. On the basis of DOE's analysis and the NRC staff's independent risk insights, DOE's technical basis to exclude the feature, event, and process from the performance assessment on the basis of low consequence is reasonable.

FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields

DOE excluded the Stress Corrosion Cracking of Drip Shields feature, event, and process from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 2 (2009ab). DOE used this feature, event, and process to consider consequences of stress corrosion cracking on drip shield materials. DOE stated that the stress corrosion cracking of Titanium Grades 7 and 29 could occur when tensile stresses exceed a threshold tensile stress value of 80 percent and of 50 percent of the yield strength at a given temperature, respectively (SNL, 2007bb). DOE stated that there are four possible sources of residual tensile stresses: (i) weld induced, (ii) caused by thermal expansion (i.e., thermal loading), (iii) plasticity caused by seismic events, and (iv) produced by rockfall and drift collapse. DOE stated that an annealing process will be used to reduce weld-induced residual stresses below the threshold tensile stress {annealing by furnace heating at 593 °C ± 10 °C [1,100 °F ± 50 °F] for a minimum of 2 hours}.

DOE considered that stress corrosion cracking may occur due to residual stresses caused by seismic events or due to stresses caused by rockfall and drift collapse. Under such conditions, through-wall cracks may form on the drip shield and seepage water may flow through those cracks. DOE supplemented the screening justification in DOE Enclosure 2 (2009ab), explaining that even if stress-corrosion cracks are assumed to penetrate the drip shield plates and remain open to water flow and if drift seepage flows through the cracks and contacts the waste package during the thermal period, the potential consequences to waste isolation are insignificant. DOE provided an additional probabilistic analysis to compute the expected number of failed waste packages within 10,000 years on the basis of the assumption that (i) waste packages could be breached by localized corrosion as a result of seismic-induced residual-stress damage of the drip shield and (ii) stress corrosion cracks on the drip shield remain open for 10,000 years and seepage water flows through unplugged cracks. The probabilistic analysis in DOE Enclosure 2 (2009ab) estimated that the mean of the expected number of waste package failures due to advection through open stress corrosion cracks on drip shields is two to three orders of magnitude lower than the mean of the expected number of waste packages failed due to early failure of the drip shields or due to seismic fault displacement involving advective flow through the waste packages (the latter cases are included in the performance assessment model). DOE concluded that because the early failure drip shields and seismic fault displacement cases are not the major contributors to the mean annual dose in the performance assessment, as shown in SAR Figure 2.4-18 and DOE Enclosure 2, Section 1.2 (2009ab), the inclusion of stress corrosion cracks on the drip shields would not significantly change the results of the performance assessment.

NRC Staff's Review

The NRC staff notes that the proposed stress-relieving process conditions are consistent with recommended industry practice (ASM International, 2003aa) to reduce residual stresses. Therefore, stress corrosion cracking of the drip shield is unlikely to occur because of weld-induced residual stresses. The thermal expansion of drip shield joints may cause residual stresses; however, DOE stated that drip shield connectors are designed to allow for thermal expansion with no effect on drip shield performance up to 300 °C [572 °F]. The thermal expansion coefficient of Titanium Grades 7 and 29 is $9.2 \times 10^{-6}\ K^{-1}$ and $9.5 \times 10^{-6}\ K^{-1}$, respectively (ASM International, 1994aa). DOE's conclusion that thermal expansion will not cause any significant tensile stresses because the drip shield temperature remains below 300 °C [572 °F] is reasonable. Below this temperature, the values of the thermal expansion coefficients of Titanium Grades 7 and 29 are not significantly different and will not lead to stress corrosion cracking.

This determination is reasonable for the following reasons: (i) DOE quantified the additional number of waste packages that could fail by localized corrosion, as a consequence of seepage infiltrating the failed drip shields, following an approach consistent with the waste package localized corrosion model evaluated in TER Section 2.2.1.3.1 and the seismic consequence abstraction model evaluated in TER Section 2.2.1.3.2; (ii) DOE concluded that the additional number of failed waste packages would be less than the number of failed waste packages for the early failure and seismic fault displacement modeling cases; and (iii) given that the contribution to the total dose of these latter cases is minimal, DOE adequately concluded that the dose contribution from the additional failed waste packages would be negligible. In addition, DOE pointed out in DOE Enclosure 2, Sections 1.2 and 1.6 (2009ab) that volumetric flow through open (unplugged) cracks is expected to be smaller than the seepage flow approaching drip shields. DOE's conclusion on the flow reduction is reasonable because (i) openings can act as capillary barriers to seep water under unsaturated conditions and (ii) DOE provided experimental evidence for the flow reduction through cracks in DOE Enclosure 2, Figure 1 (2009ab). DOE provided the same justifications to also exclude FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and the NRC staff also notes the technical basis is adequate for this other feature, event, and process. In summary, DOE's technical basis to exclude both FEPs 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields, and 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, on the basis of low consequence is reasonable.

FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields

As identified in the conclusion of the screening justification (technical basis) for this feature, event, and process (SNL, 2008ab), DOE excluded the Localized Corrosion of Drip Shields feature, event, and process from the performance assessment model on the basis of low consequence. DOE used this FEP to consider consequences of localized corrosion on drip shields. DOE stated that it evaluated Titanium Grade 7 over all the anticipated ranges of pH, chloride concentration, and temperature relevant to the proposed repository. On the basis of available information, DOE concluded that localized corrosion of Titanium Grade 7 is not expected to occur. Literature results suggest that the presence of fluoride ions can enhance the general corrosion rate of titanium alloys and possibly lead to localized corrosion. DOE stated it examined localized corrosion of titanium alloys in fluoride-containing solutions and concluded that these types of solutions would rarely occur and that low fluoride concentration in combination with expected inhibiting species (such as nitrate, carbonate, and sulfate) is unlikely to lead to localized corrosion (SNL, 2008ab). DOE noted that long-term corrosion tests of titanium alloys in repository-relevant environments up to 5 years did not indicate any evidence of localized corrosion. DOE acknowledged that data on Titanium Grade 29 are sparse and that it is less resistant to localized corrosion. DOE, therefore, postulated that localized corrosion may initiate on Titanium Grade 29. In other words, DOE stated that existing information on localized corrosion on Titanium Grade 29 is not sufficient to rule out this process or support a notion that localized corrosion is of low probability. DOE noted that the majority of the Titanium Grade 29 components except the side framework, however, would be located underneath the Titanium Grade 7 plates and would be exposed to benign environments. Therefore, DOE concluded that the drip shield could experience localized corrosion only on the side framework. However, if these side frameworks collapsed, DOE concluded that the drip shield would continue to function and protect the waste package against seepage through the Titanium Grade 7 plates (SNL, 2008ab). Therefore, DOE excluded the feature, event, and process on the basis of low consequence.

NRC Staff's Review

The NRC staff reviewed the technical basis (SNL, 2008ab) and supporting information provided by DOE [BSC (2004as); DOE Enclosure 6 (2009ab)]. The NRC staff examined DOE's model assumptions and model support in the area related to localized corrosion of the drip shield. The NRC staff notes that the 2.5- and 5-year testing DOE conducted indicates that the possibility for localized corrosion of the drip shield is small in the potential repository environment. The NRC staff performed independent analyses that indicated the concentration of fluoride, which at higher levels increases the localized corrosion susceptibility, would not likely achieve high levels in the proposed repository (Lin, et al., 2003aa; Pabalan, 2010aa). The independent analyses indicate that fluoride precipitates with common chemicals in the groundwater limiting the concentration of free fluoride ions in the water. Independent localized corrosion analyses of Titanium Grade 7 support the notion that localized corrosion of Titanium Grade 7 is not likely under repository conditions (Brossia and Cragnolino, 2000aa). Further NRC staff's evaluation of fluoride effects and long-term immersion tests are provided in TER Section 2.2.1.3.1.3.1.1.

Measurements of hydrogen absorption described in DOE's information in DOE Enclosure 6 (2009ab) and literature information (e.g., Okada, 1983aa) imply a state of passivity. NRC staff notes that the passivity of titanium and titanium alloyed with platinum or nickel is likely to be preserved, even in acid solutions with pH as low as 3.5 at 25 °C [77 °F] under cathodic polarization. Corrosion studies by Smailos, et al. (1992aa) on titanium alloyed with 0.17 percent palladium did not show localized corrosion in German rock salt repository environments under gamma radiation and temperatures ranging from 90 to 200 °C [194 to 392 °F]. In other studies by the same group, the metallic samples were subjected to adhering salts and corrosion products without significant corrosion affecting the titanium alloys (Smailos and Köster, 1987aa). NRC staff conducted corrosion tests in concentrated chloride solutions at 95 °C [203 °F] of Titanium Grade 7 galvanically coupled with Alloy 22 to form a crevice and found no indication of localized or galvanic corrosion of Titanium Grade 7 (He, et al., 2007ab). Therefore, DOE's technical basis to exclude localized corrosion of Titanium Grade 7 from the performance assessment is reasonable.

The NRC staff, in TER Section 2.2.1.3.2, evaluated the ability of the drip shield to maintain its seepage barrier function if the side framework, made of Titanium Grade 29, buckled. On the basis of that evaluation, the following DOE conclusions are reasonable: (i) the drip shield would continue to function and protect the waste package against seepage through the Titanium Grade 7 plates and (ii) localized corrosion of the drip shield would not have a significant effect on dose calculations. Thus, DOE's technical basis for exclusion of the Localized Corrosion of Drip Shields feature, event, and process is reasonable.

FEP 2.1.03.04.0B, Hydride Cracking of Drip Shields

DOE excluded Hydride Cracking of Drip Shields from the performance assessment model on the basis of low probability (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 1 (2009cb). According to DOE's definition, this feature, event, and process refers to the absorption of hydrogen into the titanium drip shield materials to form mechanically weak hydrides, which could lead to the formation of cracks. DOE noted that hydrogen absorption in titanium alloys could occur under repository conditions. DOE evaluated hydride cracking by developing a model where hydrogen-induced cracking is assumed to occur if the absorbed hydrogen resulting from general corrosion of the drip shield into Titanium Grades 7 and 29 exceeds a critical hydrogen concentration (SNL, 2008ab). DOE estimated that the amount of hydrogen uptake in 10,000 years would be below this critical hydrogen concentration. DOE tracked the drip shield materials and thickness in SNL Design Control Parameter 07-04, Table 7-5 (2008ad).

DOE also evaluated uphill diffusion along Titanium Grade 29 to Grade 7 welds, which could lead to locally elevated hydrogen concentrations near the welds. DOE concluded in DOE Enclosure 8 (2009ab) that the use of a filler metal (Titanium Grade 28) with a composition comparable to both welded components would eliminate this particular issue. By using Titanium Grade 28, DOE intended to provide a gradual aluminum concentration gradient to restrict hydride formation due to hydrogen redistribution. DOE tracked the drip shield design including welds in SNL Table 7-5, Design Control Parameter 07-01 (2008ad) and the use of Titanium Grade 28 in SNL Table 7-4, Design Control Parameter 07-12 (2008ad) as weld filler material for Titanium Grade 7 to Grade 29 welds.

DOE concluded that, given the limited extent of hydrogen formation and the use of Titanium Grade 28 filler material on weld lines, Hydride Cracking of Drip Shields can be excluded from the performance assessment model (SNL, 2008ab).

NRC Staff's Review

The NRC staff reviewed the feature, event, and process screening technical basis in SNL (2008ab) and DOE Enclosures 3–8 (2009ab). The NRC staff analyzed DOE's model assumptions and model support in areas related to hydride cracking induced by hydrogen absorption resulting from general corrosion of the drip shield and hydrogen diffusion along dissimilar titanium welds. From this review, the NRC staff noted that the critical hydrogen concentrations DOE assumed to lead to fast fracture were reasonable. Although delayed hydride cracking is possible at hydrogen concentrations as low as 30 ppm in Titanium Grade 5 steel, the applied stress intensification for the delayed hydride cracking is near to the fracture toughness limit, as described in DOE Enclosure 3 (2009ab). DOE described that palladium and ruthenium played a beneficial role by increasing the critical hydrogen concentration value and decreasing the hydrogen absorption. This is reasonable because independent literature data indicate that palladium and ruthenium can increase the critical hydrogen concentration and because the repository is predicted to be an oxic environment, as outlined in DOE Enclosure 4 (2009ab). DOE's assessment of hydrogen absorption efficiency, as indentified in DOE Enclosure 6 (2009ab), is reasonable because the experimental condition used to test hydrogen absorption bounds the range of conditions important to this mode of degradation. DOE provided distributions of hydrogen in titanium due to uphill diffusion by a stress gradient in DOE Enclosure 7 (2009ab) and due to uphill diffusion by aluminum concentration in SNL (2008ab). The NRC staff noted those hydrogen distributions are reasonable on the basis of the analysis of DOE's assumptions. Furthermore, the NRC staff developed an uphill diffusion model (Mintz and He, 2009aa), applied the model to potential repository conditions to examine the hydrogen concentration around the weld zones, and noted that hydrogen concentrations would be minimal.

DOE excluded Hydride Cracking of Drip Shields from the performance assessment model on the basis of low probability (SNL, 2008ab). DOE updated the technical basis to show that the probability of hydride cracking of drip shields is less than 10^{-4} in 10,000 years, as detailed in DOE Enclosure 1 (2009cb). DOE described that even with a high corrosion rate at a probability level of 2.5×10^{-5} (applied for 10,000 years), the hydrogen concentration would be below the critical hydrogen concentration for hydride cracking. The exclusion of the feature, event, and process on the basis of low probability is reasonable because DOE (i) considered high corrosion rates to overestimate the amount of hydrogen produced from the general corrosion process and (ii) concluded that the hydrogen concentration would not suffice to induce hydride cracking on the drip shield plate and frame. DOE's low probability conclusion on the basis of the high corrosion rate analysis and the selection and control of titanium alloy material and weld filler metal are reasonable. Therefore, DOE's technical basis to exclude hydride cracking of the drip shields from the performance assessment model on the basis of low probability is reasonable.

FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield

DOE excluded the Advection of Liquids and Solids Through Cracks in the Drip Shield from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 2 (2009ab). According to DOE's definition of the feature, event, and process, if cracks develop on the drip shield, water could flow through those cracks and contact the waste package. DOE presented technical reasons for excluding the potential of advective flow of water through cracks in a drip shield. These involved

(i) Creep/stress relaxation in a drip shield (of Titanium Grade 7) could limit the development and penetration of stress corrosion cracks

(ii) A small damaged area (less than 0.5 percent) on the drip shield surface from seismic-induced rockfall could limit the surface area available for advective flow of seepage water

(iii) A low chance of large rockfall from lithophysal rock zone above the drip shield could cause sufficient stress corrosion cracks and denting of a drip shield

(iv) A low chance of large rock-block falls from the nonlithophysal rock zone above the drip shield could occur due to low probability of seismic events of sufficient magnitude

(v) Potential filling and plugging of stress corrosion cracks by mineral precipitates and corrosion products could potentially limit the advective flow of water through a drip shield

(vi) A low chance of perfect alignment of tight and tortuous cracks on a drip shield surface could occur with impinging seepage drips from the drift wall

(vii) In the absence of drip shields, in less than 10 percent of the waste packages localized corrosion would be initiated [SNL Appendix O (2008ag)]

(viii) If leakage through a crack-damaged drip shield caused a localized corrosion of the waste package, only a small flux {4 mL/yr [0.244 in^3/yr]} would directly flow into the waste package, which would be insignificant from the repository performance standpoint (SNL, 2008ab)

Therefore, DOE excluded the feature, event, and process from the performance assessment model on the basis of low consequence.

DOE supplemented the screening justification with DOE Enclosure 2 (2009ab) and described that potential consequences to waste isolation are insignificant even if stress corrosion cracks are assumed to penetrate the drip shield plates and remain open to water flow and if drift seepage flows through the cracks and contacts the waste package during the thermal period. DOE provided an additional probabilistic analysis in DOE Enclosure 2 (2009ab) to compute the expected number of failed waste packages within 10,000 years on the basis of the assumption that (i) waste packages could be breached by localized corrosion as a result of seismic-induced, residual-stress damage to the drip shields and (ii) stress corrosion cracks on the drip shield remain open for 10,000 years and seepage water flows through unplugged stress corrosion cracks. DOE concluded that the additional number of failed waste packages would be too small to change dose estimates. In addition, DOE stated in DOE Enclosure 2, Sections 1.2 and 1.6 (2009ab) that volumetric flow through open (unplugged) cracks is expected to be smaller than volumetric seepage approaching drip shields and provided experimental evidence in DOE Enclosure 2, Figure 1 (2009ab) to support this statement. The technical basis for exclusion of the feature, event, and process is reasonable, for reasons provided under FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields, in this TER. In summary, DOE's technical basis to exclude the feature, event, and process on the basis of low consequence is reasonable.

FEP 2.1.06.04.0A, Flow Through Rock Reinforcement Materials in the Engineered Barrier System

DOE excluded Flow Through Rock Reinforcement Materials in the Engineered Barrier System on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosures 2 and 7 (2009cb). As defined by DOE, this feature, event, and process addresses the potential of groundwater flow to occur through the ground support materials such as wire mesh, rock bolts, grout, and liner. This feature, event, and process also evaluates the potential for ground support or its degradation products to enhance or decrease seepage (groundwater flow) into emplacement drifts, or to divert water flow within the drifts. In the performance assessment model, DOE assumes that seepage is not affected by any rock reinforcement materials. For boreholes, FEPs 1.1.01.01.0B, Influx Through Holes Drilled in Drift Wall or Crown, and 2.1.06.04.0A, as defined by DOE, cover similar processes and features because open space will be present in boreholes regardless of whether rock bolts degrade.

DOE stated plans to employ friction-type carbon steel rock bolts with plates for use as temporary ground support during construction of the emplacement drifts, to be left in place between the rock and the permanent (Bernold-type sheets) ground support shown in SNL Design Parameter Number 01-15 (2008ad). DOE stated in the screening justification that the seepage model indicates the presence of rock bolts does not lead to significant seepage enhancement. DOE supported this conclusion by assuming that (i) an open borehole without grout acts as a capillary barrier to unsaturated flow; (ii) a cross-sectional area of the rock bolt borehole, onto which flow may be incident, is small; and (iii) water that may have seeped into the borehole can imbibe back into the rock along its length (assumptions also related to FEP 1.1.01.01.0B, Influx Through Holes Drilled in Drift Wall or Crown). In addition, DOE indicated that the Bernold-type sheets, which are bolted to the drift walls and roof, may divert seepage. However, DOE stated that this diversion may be limited as these sheets will be perforated and the supporting rock bolts will degrade as outlined in SNL Design Parameter 01-16 (2008ad). Therefore, DOE chose not to take credit for seepage diversion by the Bernold-type liner sheets for the period before the liner would fully corrode.

DOE stated that neither the rock bolts used as temporary ground support nor those holding the Bernold-type sheets will have a significant effect on the seepage flow rate. DOE also noted that the ground support system is expected to degrade as a result of drift degradation (BSC, 2004al). Therefore, DOE described that excluding the temporary ground support in the representation of seepage in the performance assessment model is a realistic representation of the system with respect to groundwater flow into the drift. DOE thus excluded Flow Through Rock Reinforcement Materials in the Engineered Barrier System from the performance assessment model.

The NRC staff‘s evaluation for FEP 1.1.01.01.0B, Influx Through Holes Drilled In Drift Wall or Crown, discussed previously in this TER, also applies to the rock bolt aspect of FEP 2.1.06.04.0A. The basis for excluding the feature, event, and process in DOE Enclosures 2 and 7 (2009cb) included the function of the drip shield, the effect on seepage rates caused by vapor flux, and the uncertainty of capillary diversion in boreholes. Thus, for boreholes used for rock bolts, DOE provided a reasonable technical basis for excluding the feature, event, and process on the basis of low consequence (also see discussion under FEP 1.1.01.01.0B, Influx Through Holes Drilled in Drift Wall or Crown). For the Bernold-type sheets, DOE’s view that the water diversion capability of these engineered components should be neglected is reasonable because they may only partially divert seeping water from contacting the drip shield or waste package until a time when the liners would fully corrode. Therefore, DOE’s technical basis to exclude the FEP Flow Through Rock Reinforcement Materials in the Engineered Barrier System from the performance assessment on the basis of low consequence is reasonable.

FEP 2.1.06.06.0B, Oxygen Embrittlement of Drip Shields

DOE excluded Oxygen Embrittlement of Drip Shields from the performance assessment model on the basis of low probability (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 9 (2009ab). DOE used this feature, event, and process to refer to oxygen embrittlement as a potential failure mechanism for the drip shields, resulting from diffusion of oxygen in titanium alloys. According to DOE, oxygen embrittlement may affect mechanical properties of the drip shield materials. DOE’s screening justification considered oxygen diffusion data at 300 °C [572 °F] by Rogers, et al. (1988aa), who used single crystal, pure titanium to estimate the oxygen lattice diffusion coefficient in alpha-phase titanium. DOE considered oxygen lattice diffusion data to estimate oxygen penetration depth for Titanium Grade 7 and concluded that any penetration depth would be minimal in 10,000 years. DOE used 300 °C [572 °F] as the bounding drip shield temperature for analysis of oxygen embrittlement. DOE stated that the 300°C [572 °F] temperature selected for the analysis could only be exceeded in the case of a drift collapse and that the probability of conditions leading to drip shields exceeding 300 °C [572 °F] is about 1 in 10,000 within the first 10,000 years of disposal. Therefore, because of this low probability and minimal oxygen penetration that may occur in 10,000 years, oxygen embrittlement of the drip shields was deemed unlikely and this process was excluded from the performance assessment model on the basis of low probability.

The NRC staff reviewed the screening rationale and DOE’s conclusion in DOE Enclosure 9 (2009ab) and SNL (2008ab) that the penetration depth would be minimal in 10,000 years. DOE cited the work of Liu and Welsch (1988aa) to support the statement that for alpha-phase titanium (e.g., Titanium Grade 7), oxygen diffusivity is independent of the form of the material (single crystal of polycrystalline) and that mass transport of oxygen is controlled by bulk diffusion through the alpha matrix (which is a slow process). For an alpha-beta alloy such as Titanium Grade 29, DOE cited additional work by Liu and Welsch (1988ab) to support the statement that the properties of the alpha phase solely control the overall oxygen embrittlement. Therefore, on the basis of work DOE cited, the NRC staff notes that the use of the bulk diffusivity of oxygen through the alpha matrix for the embrittlement calculation for both alpha (Titanium Grade 7) and alpha-beta (Titanium Grade 29) alloys is reasonable. Also, the NRC staff notes that DOE overestimated the oxygen penetration depth due to the assumption of a constant temperature of 300 °C [572 °F] in analyses in DOE Enclosure 9 (2009ab) and SNL (2008ab). DOE's results of temperature computations for the drift-collapsed case indicate average waste package temperatures below 300 °C [572 °F] (SAR Figure 2.3.4-98), implying drip shield temperatures also below 300 °C [572 °F]. The NRC staff evaluates system temperature computations in TER Section 2.2.1.3.6. DOE's conclusion that oxygen penetration would be minimal is reasonable on the basis of DOE's computations at 300 °C [572 °F], which indicate that oxygen embrittlement of the drip shield is unlikely. Therefore, DOE's technical basis to exclude the feature, event, and process on the basis of low probability is reasonable.

FEP 2.1.07.02.0A, Drift Collapse

As defined by DOE, this feature, event, and process considered nonseismic drift collapse; specifically, the degradation of emplacement drifts that may result from the combination of excavation-induced rock stress and thermal loading in the absence of significant seismic events. DOE considered seismically induced drift collapse as a separate feature, event, and process that was included in its performance assessment evaluation. Seismically induced drift collapse is reviewed as a model abstraction in TER Section 2.2.1.3.2.5 and, hence, is not addressed in this subsection.

DOE stated that degradation of waste emplacement drifts can occur from stresses that exceed the strength of the rock mass surrounding the drift. These stresses are attributed to several causes. One cause is excavation-induced stresses that are superimposed on the *in-situ* (geostatic) stresses soon after the drifts are constructed. Another cause is thermally generated stresses. After waste emplacement, thermal stresses develop in the rocks from heat generated through radioactive decay of the emplaced waste. In addition, rocks under the influence of combined mechanical and thermal stresses may experience a gradual weakening with time. Rocks can be expected to fail when any of the stresses, individually or in combination, exceed the rock strength. Such failures can cause a gradual accumulation of rubble on and around the engineered barriers as a result of a continuing but slow process of rockfall. Alternatively, rocks above the emplacement drift could collapse due to a combination of all the stresses that exceed the strength of the rock mass. (In this context, "rock mass strength" refers to the strength of the larger volume of rock around the waste emplacement drift whose behavior under stress is controlled by the presence of fractures, discontinuities, and cavities, as opposed to the strength of a small-sized intact rock core sample used in laboratory testing.) Both the gradual accumulation of rubble and instantaneous collapses of massive rocks may have undesirable consequences on the performance of the engineered barriers, depending on their magnitude (e.g., small, medium, or large amounts of rockfall).

DOE characterized the rock properties and applied several analytical tools and numerical models to assess the long-term behavior of rocks under coupled natural and repository-induced processes as a function of time. Uncertainties in the long-term behavior of the rocks were incorporated in the analyses.

DOE stated that drift degradation could occur rapidly if the stress change is large enough to cause instantaneous rock failure or gradually if the stress change is too small to cause rapid failure but large enough to weaken the rock with time. DOE summarized its basis for excluding drift collapse in SNL (2008ab). DOE provided detailed supporting analyses in BSC (2004al). DOE also supplemented its technical basis for exclusion of the feature, event, and process in DOE Enclosures 1–8 (2009ae), DOE Enclosure 1 (2009cd), DOE Enclosure 1–2 (2009ce), DOE Enclosure 1 (2009cf), DOE Enclosure 1 (2009cg), and DOE Enclosure 1 (2009ch). DOE concluded that nonseismic drift degradation would cause only minor, localized rockfall that results in insignificant impact on the thermal and hydrologic conditions of the drift and minimal consequences to the engineered barrier system components.

DOE addressed the analytic models it developed for lithophysal and nonlithophysal rock types in BSC Sections 6.3 and 6.4 (2004al). The predominant surroundings of the emplacement drifts consist of lithophysal rocks.

DOE evaluated effects of postexcavation and thermal stresses in lithophysal rocks using a two-dimensional, drift-scale discontinuum Voronoi block model when applying the UDEC code (Itasca International, Inc., 2004ac, as described in BSC, 2004al) to analyze the mechanical behavior of drifts for five rock-strength categories of lithophysal rock, as detailed in BSC Section 6.4.2.1 (2004al). UDEC is a computer code used internationally by the rock mechanics and mining industries both as a research tool and a design tool. There are numerous, extensively peer-reviewed scientific papers and refereed journal articles on the use of UDEC code. In its implementation of the UDEC code, DOE chose the discontinuum Voronoi approach because the model allows computation of both the time-dependent stress-strain response of rock to thermal loading and the dynamic response of the rock mass under seismic events that can lead to rockfall. The processes considered within the Voronoi domain are gravitational stresses, excavation-induced stresses, thermally induced stresses, and time-dependent strength degradation. Under the defined model domain and boundary conditions, the UDEC–Voronoi model is used to calculate mechanical response of the Voronoi domain to a set of imported temperature distributions that are updated at 45 discreet timesteps to cover a 10,000-year period following repository closure.

NRC Staff's Review

The NRC staff used YMRP Section 2.2.1.2, Review Method 2, to evaluate DOE's analyses. Specifically, the NRC staff evaluated DOE's analyses and calculations supporting its screening basis and its use of bounding or representative estimates for the consequences. The NRC staff performed independent calculations using analytical tools and numerical models to scope potential issues and to verify or confirm DOE's conclusions. In evaluating DOE's technical basis for excluding the feature, event, and process, the NRC staff also considered its independent analysis (Ofoegbu, et al., 2007aa).

DOE summarized its basis for excluding drift collapse in SNL (2008ab). DOE provided detailed supporting analyses in BSC (2004al). DOE also supplemented its technical basis in DOE Enclosures 1–8 (2009ae), DOE Enclosure 1 (2009cd), DOE Enclosure 1–2 (2009ce), DOE Enclosure 1 (2009cf), DOE Enclosure 1 (2009cg), and DOE Enclosure 1 (2009ch). On the basis of the NRC staff's review of that information, the NRC staff focused its detailed review on the following aspects of the DOE model that are important in estimating rock response to postexcavation and thermal stresses:

- Characterization of rock mechanical and thermal properties
- Model domain and boundary conditions
- Initial stress state and rock temperature inputs
- Block size and shape in the Voronoi domain
- Model calibration
- Model results (extent and timing of rockfall)
- Model support
- Treatment of time-dependent failure
- Alternative conceptual models

The following subsections summarize DOE's approach and the NRC staff's evaluations for each of these aspects of the DOE technical basis and conclusions.

Characterization of Rock Mechanical and Thermal Properties

The Yucca Mountain site-specific geologic characterization of the rock units was accomplished by geologic mapping of the Topopah Spring Tuff, which was identified as the host rock. The Topopah Spring Tuff includes both lithophysal and nonlithophysal rock units. Approximately 15 percent of the emplacement block consists of nonlithophysal rocks that are hard, strong, fractured masses. The remaining 85 percent of the repository block consists of lithophysal rocks that are more deformable with lower compressive strength than the nonlithophysal units. Different rockfall analysis methods were used for these two rock types (SNL, 2008ab). Because the predominant surroundings of the emplacement drifts consist of lithophysal rocks, the NRC staff focused its review of DOE's technical basis for excluding drift collapse on DOE's modeling of lithophysal rocks.

DOE performed laboratory and *in-situ* testing to derive the mechanical and thermal properties of the lithophysal rocks used in its analysis. The mechanical and physical properties included elastic moduli, unconfined and triaxial compressive strength, tensile strength, density, porosity, normal and shear stiffness, and shear strength. The geometric rock fracture properties included dip and dip direction, spacing, length, surface roughness, and microstructure. These properties were obtained from laboratory tests of small- and large-diameter cores. The rock mass strength properties were established by *in-situ* measurements. Thermal properties measured in the laboratory and *in situ* included thermal conductivity, thermal expansion coefficient, and heat capacity.

DOE studied the time dependence of intact rock strength. These parameters (e.g., static-fatigue data at given environmental conditions of moisture and temperature) were used in the time-dependent drift degradation calculations to define the rate of strength decay as a function of stress state. The effects of sample size, anisotropy, and sample saturation were studied. DOE showed that the unconfined compressive strength decreases with increases in sample size. DOE reported a maximum anisotropy of 10 percent in the average matrix moduli, which, according to DOE, is a second-order effect compared to the effect of lithophysae and fracturing on moduli and strength, as described in BSC Appendix E (2004al). DOE also found that the variability in elastic and strength properties is not a function of lateral or vertical position within the repository host horizon, but primarily is a function of porosity of the samples (BSC, 2004al). DOE accounted for uncertainty in modeling the time dependence of intact rock strength by bounding the range of rock mechanical properties as a function of porosity, temperature, and saturation.

DOE stated that the major difference in fracture characteristics between the nonlithophysal and the lithophysal rocks is the trace, or fracture length. For the nonlithophysal rocks, the average fracture length is greater than or equal to 1 m [3.28 ft]; for the lithophysal rocks, fracture lengths average less than 1 m [3.28 ft]. The abundant small-scale fractures in the lithophysal rocks result in the weaker nature of this rock, and the potential failure will be in a raveling mode that results in generally small block sizes. The major fracture differences between lithophysal and nonlithophysal rocks influenced DOE's choice for the numerical codes used for the drift stability analyses.

The thermal properties of the lithophysal rocks were derived from laboratory and field measurements (BSC, 2004al). To account for the uncertainty in the thermal properties, DOE used a coefficient for intact rock in the thermal-mechanical rockfall analysis, which DOE concludes leads to larger and, thus, conservative, thermally induced stresses as shown in BSC Appendix E (2004al).

A commercial discontinuum numerical modeling particle flow code PFC2D [Itasca International, Inc. (2004ab), as described in BSC (2004al)] was used to evaluate the effect of lithophysal size, shape, and distribution on the variability of the mechanical properties. This numerical analysis simulates the basic deformation and failure response mechanism of lithophysal tuff (BSC, 2004al). Bounding ranges for mechanical properties were established using this method. To determine rock-strength characteristics, DOE combined the PFC modeling results with the laboratory test data. The unconfined compressive strength was plotted as a function of the Young's modulus in BSC Appendix E (2004al). The analysis identified a lower bound strength cutoff at 10 MPa for lithophysal rocks. The sensitivity studies using stress analysis models found that instability would be expected to occur if the *in-situ* rock strength was below about 10 MPa [BSC Appendix E (2004al)]. DOE supported this conclusion with field observations from the existing Exploratory Studies Facility and the Enhanced Characterization of the Repository Block cross-drift tunnels. Thus, this strength–Young's modulus plot is used as the basis for dividing the lithophysal rocks into five strength categories for rockfall modeling. DOE stated in BSC Section 6.4.1.2 (2004al) that the lowest ranges of strength categories with porosity greater than 20 percent likely underestimate the true rock-mass strength.

NRC Staff's Review of Characterization of Rock Mechanical and Thermal Properties

The NRC staff reviewed the methods described in BSC (2004al) and notes that DOE followed standard industry practices and methods for host rock characterization. The mechanical and thermal properties of the rocks were acquired through laboratory and field tests with samples and/or sites to adequately characterize uncertainty in relevant parameters. DOE's use of numerical analyses to supplement laboratory data and field measurements is reasonable due to the practical limitations of obtaining large samples in weakly coherent lithophysal rocks. The NRC staff reviewed DOE's use of the PFC2D modeling code [Itasca International, Inc. (2004ab) as described in BSC (2004al)] to simulate deformation and failure response mechanisms of lithophysal rocks. This modeling approach is reasonable, because DOE applied standard industry practices and qualified methods, as detailed in BSC Section 3 (2004al), for characterizing the rock properties. The data uncertainty and their natural variability were captured and used in the numerical modeling to analyze drift stability.

In a request for additional information, the NRC staff asked DOE how uncertainties in stress-strain relationships for lithophysal rocks were characterized by the number of laboratory tests conducted, as outlined in DOE Enclosure 1 (2009ce). DOE Enclosure 1 (2009ce) provided additional details on the stress-strain relationships for lithophysal rocks, which showed that the tested rocks have a more ductile response (i.e., less prone to failure at peak stress) than the simulated rock mass in the UDEC [Itasca International, Inc. (2004ac), as described in BSC (2004al)] models. The NRC staff reviewed this information and noted that uncertainties in the stress-strain relationships for lithophysal rocks would not affect the model results significantly, because DOE represents the modeled rock mass as more prone to brittle failure than the actual rock mass.

Model Domain and Boundary Conditions

DOE used the NUFT thermo-hydrology continuum model (Lawrence Livermore National Laboratory, 1998aa) as described in BSC (2004al) to simulate the two-dimensional, drift-scale, thermal-hydrologic behavior and the FLAC 2-D continuum code [Itasca International, Inc. (2004aa), as described in BSC (2004al)] to calculate thermally induced stresses. DOE used the UDEC 2-D discontinuum computer code [Itasca International, Inc. (2004ac), as described in BSC (2004al)] for the drift stability analysis in lithophysal rock because the discontinuum approach best represented the highly fractured character of the lithophysal rock. In the UDEC lithophysal rockfall model, the region around the drift, where inelastic deformation is expected to occur, is discretized into discrete blocks using a mathematical relationship called a Voronoi tessellation. The Voronoi model was used to represent the random orientations of the rock blocks. DOE obtained the specified average dimension from the characterization of the rocks. In the UDEC model, the Voronoi block domain around the drift is bounded by large, continuous blocks with elastic properties. The temperature-time history from NUFT was mapped onto the UDEC grid blocks. To assess the repository edge effects and topographic influences on the temperature and thermal stress distributions, DOE performed coupled, three-dimensional (multiple drifts), regional- and drift-scale calculations using FLAC3D [three-dimensional continuum code; Itasca International, Inc. (2004aa), as described in BSC (2004al)].

A coupled three-dimensional regional- and drift-scale thermal-mechanical calculation was conducted to support the two-dimensional drift-scale calculation. The three-dimensional analysis was performed in two steps. First the regional scale thermal-mechanical calculation was used to calculate the temperature and stress changes on the entire mountain. Then the detailed local scale [also called large scale in BSC Appendix C (2004al)] thermal-mechanical analysis was performed such that the boundary conditions for temperature and stresses were obtained from the regional-scale calculation, as outlined in BSC Section 6.2 (2004al).

The temperatures and stresses calculated by the drift-scale model (NUFT-FLAC results), in which simplified rigid boundary conditions (zero displacement) are assumed for the vertical and bottom boundary planes, were compared with the coupled, three-dimensional, regional- and drift-scale model (FLAC3D results). The comparison showed that the simplified rigid boundary condition used in the two-dimensional drift-scale model resulted in higher horizontal stresses compared to the three-dimensional regional model, especially at the repository edge where the confinement and temperatures are less than in the middle of the repository. Thus, DOE concluded in BSC Section 6.2 (2004al) that the two-dimensional model provides conservative conditions for use in the drift degradation analyses.

In the drift-scale calculation, a symmetric boundary condition is applied on a vertical plane halfway between the emplacement drifts. This modeling technique results in zero displacements (i.e., full confinement) perpendicular to the boundary and zero heat flux across the boundary, as described in BSC Section 6.2 (2004al). These boundaries account for the symmetry of mechanical behavior on either side of the vertical plane between parallel drifts, assuming that parallel drifts undergo similar thermal loads. DOE compared the stresses calculated using these boundary conditions to stresses from the coupled regional and drift-scale calculations. On the basis of this comparison, DOE concluded in BSC Section 6.2 (2004al) the vertical boundary conditions in the UDEC–Voronoi model overestimate the thermal stress for drifts near the margins of the repository area.

The bottom boundary of the UDEC–Voronoi model is also fixed, which treats the underlying Earth's crust as a rigid body. The top of the model is assigned a constant-stress boundary condition, fixed at the estimated vertical *in-situ* stress at a 300-m [984-ft] depth. In BSC Appendix W (2004al), DOE provided sensitivity analyses that show the calculated stresses at the drift walls are insensitive to extension of the model boundaries beyond the distances considered in the current models.

NRC Staff's Review of Model Domain and Boundary Conditions

The NRC staff reviewed the details of DOE's numerical models and related calculations used to determine boundary conditions. The computer codes DOE used in the thermal-mechanical boundary calculations are well tested and appropriate for their intended use.

To evaluate the adequacy of DOE's selection of boundary conditions, the NRC staff conducted confirmatory thermal-mechanical calculations (Cao, 2010aa) using analytical and finite element methods (Abaqus computer code; Dassault Systèmes Simulia Corp, 2009aa) for a single heated drift. The NRC staff used the analytical solution of Kirsch (Jaeger, et al., 2007aa) for a circular tunnel as the sum of *in-situ* stress and excavation-induced stress and then added the thermal stress, which was calculated by solving the Laplace equation assuming symmetrical temperature distribution in the radial direction. Using this approach, the NRC staff calculated similar stress values at the crown and sidewall areas, as DOE analyzed in BSC Figures 6-31 to 6-33 (2004al), when boundary conditions similar to the DOE UDEC–Voronoi model were used.

The NRC staff's confirmatory calculation with the rigid boundary condition also supports DOE's conclusion that horizontal stresses are overestimated for drifts near the edge of the repository. By using a fixed boundary condition for the UDEC–Voronoi model, DOE does not allow for potential horizontal expansion to reduce the accumulation of horizontal stress from thermal expansion of the rock. The use of fixed vertical boundaries in the DOE model is reasonable, because this assumption will not underestimate the potential effects of thermal stress on rocks surrounding the heated drifts.

The NRC staff reviewed the technical details of DOE's analyses, presented in BSC Appendix W (2004al), to determine whether the boundary conditions in DOE's model were appropriately selected. The NRC staff notes that the model boundary below the drift is located within the outer limits of the thermally disturbed zone around a drift. This implies that some component of thermal expansion may not have been fully considered in the model. However, any thermal expansion in this zone does not influence the rockfall estimates significantly, because only a small increase in rock stress would be expected for this small increment in temperature. On the basis of the sensitivity analyses DOE provided, the NRC staff notes that the magnitude of that potential component is negligible and would not significantly affect the calculated stresses near the drift. The dimensions of the DOE model domain are reasonable, because consideration of an extended region does not affect significantly the potential effects of thermal stress on rocks surrounding heated drifts.

Initial Stress State and Temperature Inputs

The DOE model assesses the preexcavation *in-situ* stresses of 7 MPa vertical and 3.5 MPa horizontal for all simulations. The vertical component represents the stress at an overburden depth of 300 m [984 ft], and the horizontal component is simplified to be 3.5 MPa on the basis of an average horizontal-to-vertical stress ratio of 0.5, as identified in BSC Section 6.3.1.1 (2004al). To obtain the postexcavation equilibrium state as the initial condition for the thermal simulations, DOE performed a quasi-static simulation in which the preexcavation stresses are applied and the model is allowed to equilibrate, as detailed in BSC Section 6.4.2.2 (2004al). Once the initial postexcavation stress state is established, spatial temperature distributions are mapped onto the model grid blocks and updated for 45 discreet timesteps as a function of time over the 10,000-year simulation period following closure. The temperature inputs as a function of time are derived from the drift-scale model using the NUFT code (Lawrence Livermore National Laboratory, 1998aa, as outlined in BSC Appendix U (2004al), and interpolated onto the UDEC model grid. The UDEC–Voronoi model then computes changes in stress state with each update in temperature input for each of the timesteps.

NRC Staff’s Review of Initial Stress State and Temperature Inputs

The NRC staff reviewed DOE’s evaluation of the initial stress state at the repository horizon. By a confirmatory hand calculation, using rock density and distance from the surface to the drift, the NRC staff reproduced and noted that DOE’s analysis of the average vertical load of 7 MPa is reasonable for the lithostatic stress at a depth of 300 m [984 ft] beneath Yucca Mountain. The NRC staff reviewed the references DOE cited in BSC Section 6.3.1.1 (2004al) regarding measurements of *in-situ* horizontal stress at Yucca Mountain. The referenced literature indicated the horizontal component of *in-situ* stress from hydraulic fracturing measurements is likely to be 1–2 MPa lower than DOE assumed. DOE’s use of 3.5 MPa is reasonable, because a 1–2 MPa overestimate in the *in-situ* horizontal stress would increase the magnitude of horizontal stress from thermal effects and, thus, overestimates the potential for rockfall.

DOE calculated the temperature inputs for the UDEC model using a detailed flow and transport code. The NRC staff performed confirmatory temperature calculations using an alternative flow and transport code (Manepally, et al., 2004aa). By comparison to NRC staff’s independent temperature calculations, the DOE temperature inputs to the UDEC model are reasonable and would not underestimate the thermal response of the heated drifts.

Block Size and Shape in the Voronoi Domain

In DOE Enclosure 2 (2009ae), DOE’s evaluation of the rock types concluded that a relatively ductile and highly jointed rock mass will fail and separate from the main body preferentially along existing discontinuities, such as fractures, joints, and intersect lithophysal cavities, and crumble. In a brittle, nonlithophysal rock mass, new fractures are expected to penetrate intact rock blocks. Thus, DOE concluded in DOE Enclosure 2 (2009ae) that thermal expansion of the Topopah Spring lower lithophysal tuff could result in movement along existing joints and deformation of lithophysal voids, whereas thermal expansion of the Topopah Spring nonlithophysal tuff could cause spalling of platy rock fragments from drift walls along newly created fractures.

To represent the lithophysal tuff, DOE used a Voronoi tessellation approach in the UDEC model (Itasca International, Inc., 2004ac) to generate a series of model elements that represent random blocks of rock surrounding the drift opening, as described in BSC Section 6.4.2.1 (2004al). The interfaces between the blocks are intended to represent the approximate spacing and random nature of existing fractures and voids in the lithophysal rock. The blocks average 30 cm [11.8 in] in diameter and are relatively uniform in size, with the largest blocks being twice the size of the smallest blocks, as outlined in DOE Enclosure 2 (2009ae). DOE concluded that an average 30-cm [11.8-in] block diameter is representative of the internal discontinuities (i.e., fractures and voids) within the lithophysal tuff. DOE conducted sensitivity analyses using average block sizes of 20 cm [7.9 in], as detailed in BSC Sections 6.4.2.3.1 and 7.6.7.1 (2004al); 10 cm [3.9 in], as outlined in DOE Enclosure 2 (2009ae); and 4 cm [1.6 in], as identified in DOE Enclosure 1, (2009ch). Although some realizations showed a small increase in the amount of fracturing and rockfall with decreasing average block size, DOE concluded these small increases are not significant with respect to the engineered barrier system performance. DOE concluded that the results of the UDEC analyses are insensitive to variations in average block size from 4 to 30 cm [1.6 to 11.8 in].

NRC Staff's Review of Block Size and Shape in the Voronoi Domain

The NRC staff reviewed DOE's technical basis used to represent lithophysal rock in the UDEC model. The NRC staff evaluated DOE's conclusion in DOE Enclosure 2 (2009ae) that yielding in heated lithophysal tuff should occur preferentially on existing structural discontinuities because the strength of the intact blocks is at least twice the strength of the discontinuous rock mass. DOE's conclusion, outlined in BSC Section 7.6.5.1 (2004al), is reasonable because fractures are distributed in a manner that rock movement associated with thermal expansion can be accommodated by slippage along a fracture path composed of coalescing "potential fractures" to form a distinct separation plane. Thus, the NRC staff reviewed how DOE's model represents yielding of the rock mass along an organized fracture network that is oriented appropriately to the applied stress.

Although DOE represents block surfaces in the Voronoi model as randomly oriented with effective blocks on the order of 30 cm [11.8 in], DOE characterized, in BSC Section 7.3.2 (2004al), the Topopah Spring lower lithophysal tuff as having primarily vertical fractures with spacing between the fractures on the order of several centimeters. DOE provided additional basis for its conclusions in DOE Enclosure 2 (2009ae), describing that the presence of lithophysal voids creates a generally isotropic rock mass. How such voids randomized the potential effects of a strongly vertical anisotropy in the rock mass was addressed in DOE Enclosure 1 (2009ch). DOE stated that visually apparent anisotropy does not affect damage and fracturing mechanics of the drift crown where the major principal stress and stress-induced fractures are normal to the subvertical fractures. This response is reasonable because at the crown area, the horizontal stress causes fracturing, and thus the rock deformation is not affected by the vertical fractures. Therefore, the model will not underestimate the magnitude of rockfall. DOE also provided observations to show the random locations and shapes of the lithophysae and the close spacing and short trace lengths of fractures indicating that a homogeneous, isotropic model provides a model of the lithophysal unit. The size of the internal structure and spacing of fractures is much smaller than the size of a drift, and therefore DOE's conclusions with respect to the drift-scale behavior of rock degradation are reasonable.

DOE Enclosure 4 (2009ae) analyses showed that the crown of the heated drifts has an overstressed zone that is approximately tens of centimeters thick. This overstressed zone is spanned by only one to two Voronoi blocks in DOE's model. The NRC staff noted that, according to DOE's analyses, a larger number of blocks might be needed to form a coherent network of surfaces to represent yielding within the rock mass BSC Section 7.6.5.1 (2004al). Because of the random distribution of block surfaces in the UDEC model, it was not clear that a coherent fracture network could form within the thin, overstressed zone. Although some block surfaces are oriented to allow yielding, these surfaces usually terminate against adjacent block boundaries that cannot yield. Thus, movement along the yielding surfaces is effectively transferred to elastic strain along nonyielding blocks within the overstressed zone. The elastic strain within the nonyielding blocks inhibits the formation of a coherent fracture network within the overstressed zone, which is necessary to represent potential yielding within the rock mass, as identified in BSC Section 7.6.5.1 (2004al). DOE addressed this issue in DOE Enclosure 1, (2009ch).

In DOE Enclosure 1 (2009ch), DOE reduced the average size of the discretized blocks to 4 cm [1.6 in]. This sensitivity analysis simulated a larger number of small-scale fractures, resulting in minor rockfall, but leading to the same depth of fracturing as the models with larger block sizes. This result is consistent with the NRC staff's confirmatory calculation (Cao, 2010aa), in which the balance between the confined rock strength and the total applied stress, thermal and *in-situ*, determines the rockfall depth. When the rockfall reaches a certain depth, where the balance is achieved, the self-arresting of rockfall is also reached. DOE's response clarified that a coherent fracture pattern forms when the block size is much smaller than the dimension of the overstressed zone. The fracturing may not be coherent if the block size is comparable to the dimension of the overstressed zone, but the failure will still be evident in the UDEC–Voronoi block model. This is true even if there are only two blocks in the zone width. DOE emphasized that the 20 to 30-cm [7.9 to 11.8-in] block sizes are appropriate because of the existing average spacing of the "preexisting" discontinuities in the rock. Thus, the rocks would result in incoherent fracture pattern with minor rockfall. On the basis of these discussions, the quantity of rockfall is not underestimated when implementing the average block size adopted in DOE analyses.

Model Calibration

In BSC Sections 7.6.3 and 7.6.4 (2004al), DOE described the approach used to calibrate the Young's modulus and unconfined compressive strength of the rock mass modeled to the expected characteristics of the lithophysal rock. Five rock-strength categories were considered in the calibration to represent the range of values for estimated Young's modulus in the lithophysal rock. For each of the five rock-strength categories considered, DOE used a mean value for unconfined compressive strength as indicated in BSC Appendix E, Figure E-13 (2004al). DOE then adjusted four Voronoi block interface properties to achieve the calibration: (i) cohesion, (ii) friction angle, (iii) normal stiffness, and (iv) shear stiffness. The calibration was repeated iteratively until the UDEC model reasonably reproduced the mean unconfined compressive strength and mean Young's modulus for each rock-strength category. Separate calibrations were performed using different values of mean block size. A 30-cm [11.8-in] average block size was used for the screening analysis. Models with average block sizes of 20 cm [7.9 in] in BSC Sections 6.4.2.3.1 and 7.6.7.1 (BSC, 2004al) and 10 cm [3.9 in] in DOE Enclosure 2 (2009ae) were developed for sensitivity analyses to ensure convergence of results.

A potentially important uncertainty in the DOE model is the representation of spatial variability in rock properties. DOE addressed this uncertainty by developing calibrated models for five different rock-strength categories, which are distinguished by different values of rock mass modulus. In conducting its calibration, DOE used the mean value of unconfined compressive strength as the calibration target for each selected value of rock mass modulus. DOE data, on the other hand, showed a range of potential values of unconfined compressive strength for a given value of rock mass modulus [BSC Appendix E, Figure E–13 (2004al)]. DOE stated (SAR p. 2.3.4-73) that a number of parametric studies were conducted in which the Young's modulus and strength parameters were varied to account for the bounding ranges of lithophysal and nonlithophysal rocks.

NRC Staff's Review of Model Calibration

The NRC staff reviewed the results of DOE's analyses using the lower bound strength and Young's modulus for rock mass Categories 2, 3, 4, and 5, as outlined in DOE Enclosure 1 (2009cd). The NRC staff compared the results of these lower bound analyses with analyses using mean values and notes that there is no significant difference in the amount of rockfall calculated using either parameter set. Thus, DOE has appropriately accounted for uncertainty in rock properties that are important in the UDEC–Voronoi model.

The NRC staff observed from DOE's analyses in DOE Enclosure 1, Table 1 (2009cd) that the amount of rockfall at the crown area is the same for lithophysal rock Categories 2 to 5 for a range of lower bound unconfined compressive strength. DOE explained that rockfall is insensitive to the change of lower bound strength because the associated change of Young's modulus, which causes the ratios between lower bound strength and Young's modulus to vary by only a few percent as shown in DOE Enclosure 1, Table 1 (2009cd). The NRC staff notes that this relationship is reasonable and that uncertainty in important rock-strength properties is appropriately considered in the UDEC–Voronoi model.

Model Results

DOE conducted extensive modeling studies to estimate the timing and extent of thermal drift degradation. The following summarizes DOE's modeling results:

- The combined *in-situ* and thermal mechanical stress reached in the drift crown is about 7 MPa for Category 1 and about 37 MPa for Category 5 lithophysal rocks, respectively, as shown in BSC Figures 6-142 and 6-144 (2004al).

- These stress values can, in some conditions, slightly exceed the unconfined compressive strength of lithophysal rock.

- The elastic stress paths cover a time range of 10,000 years' variation of temperature.

- The amount of thermally induced rockfall is small for all five categories of lithophysal rocks.

- Basic rock mechanics principles show that the potential for the thermally induced rockfall process should cease at a short distance from the drift, where the confined strength of the rock is greater than the sum of mechanical and thermal stresses.

NRC Staff's Review of Model Results

To evaluate the amount of stress likely produced during thermal heating of the rocks surrounding the drifts, the NRC staff conducted confirmatory calculations using the Abaqus continuum model (Cao, 2010aa). The results of these calculations reasonably represented the stress levels DOE calculated for different categories of lithophysal rocks. The stress level for Category 1 rocks remains below the strength of the rock. Consequently, rockfall is not expected to occur for Category 1 rocks. The NRC staff's analyses confirmed that for lithophysal rock Categories 2–5, compressive stresses in some parts of the drift wall can exceed the unconfined compressive strength of the rock mass.

The NRC staff reviewed the results of UDEC–Voronoi simulations that showed limited amounts of rockfall occur when an overstressed zone (i.e., where horizontal compressive stress may exceed the unconfined compressive strength of the rock) develops in the drift wall. The NRC staff's confirmatory calculations (Cao, 2010aa) showed that an overstressed zone is expected to occur within the first tens of centimeters of the drift wall, which is comparable to the depths calculated in DOE Enclosure 1 (2009cd).

The NRC staff focused on determining the reasonableness of DOE model results that showed only limited amounts of rockfall can occur in the overstressed zone. The UDEC–Voronoi model relies on accommodating some degree of rock stress by movement along the interfaces between adjacent blocks. When the applied stress exceeds the ability of the blocks to move, the interfaces can fail and blocks can separate from the modeled rock mass. The NRC staff questioned whether the block size in the UDEC–Voronoi model was small enough to capture a through-going failure of adjacent blocks within the narrow overstressed zone. In response, DOE provided supplemental analyses in DOE Enclosure 1 (2009ch) that demonstrated failure patterns in models with 4-cm [1.6-in] average block sizes are comparable to failure patterns in models with larger block sizes. The NRC staff reviewed these results and notes that the UDEC–Voronoi model appears capable of representing block failure in the overstressed zone for average block sizes that range from 4 to 30 cm [1.6 to 11.8 in].

The NRC staff evaluated information provided in DOE Enclosure 5 (2009ae) that further explained why failure of the rock mass is expected only in a thin zone around the drift walls. The NRC staff's confirmatory calculations (Cao, 2010aa) verified that local block failure at the drift wall should cause stresses to accumulate farther outward from the drift surface, where the rock mass is confined and more resistant to failure. Thus, as blocks fail along the overstressed zone on the drift wall, the stress concentration is expected to shift outward from the drift wall into the more confined area where the strength is higher. This process reasonably explains why only a limited amount of rockfall is expected from thermal-mechanical effects on lithophysal rocks.

In reviewing the UDEC–Voronoi model results, the NRC staff observed that some blocks appear to maintain cohesion with adjoining blocks when the interface between the blocks is in an apparent state of failure. DOE provided additional information in DOE Enclosure 2 (2009ce) to show that although some part of the interface failed, some other parts of the interface retained sufficient strength to support the hanging block. Blocks also can remain intact if the geometry of adjacent blocks continues to support the block after an interface has failed, as shown in DOE Enclosure 2 (2009ce). The NRC staff reviewed the results of DOE's calculation, which showed that the UDEC code appropriately analyzes cohesion within adjoining blocks (i.e., beam support), and notes that the UDEC–Voronoi model appropriately calculates limited amounts of rockfall as occurring from the overstressed zone in the drift walls.

The NRC staff conducted confirmatory calculations to support the review of DOE's calculation of limited amounts of rockfall from the thermal-mechanical effects of waste emplacement (Cao, 2010aa). For example, Kaiser, et al. (2000aa) showed that rocks that are subject to spallation (i.e., an assumed mode of failure from thermal-mechanical effects on the drift wall) typically form inverted v-shaped notches in the drift crown that limit the extent of rockfall. The NRC staff used the Abaqus computer program to evaluate the differences in stress conditions between a circular drift and a circular drift with an upside-down v-shaped notch. The NRC staff's analysis confirmed that the presence of the v-shaped notch would lead to self-arrest of spallation, because the tangential stresses on both sides of the notch were released because of confinement loss or because there was room for physical expansion for stress release. The v-shaped notch had a dimension approximately equal to the depth of the overstressed zone above the crown. Tangential stresses in the plane of the tunnel were compared to the confined or unconfined rock strength, as appropriate. The calculation shows that even if the thermo-elastic stresses exceed the unconfined compressive strength of the rock, as the failure zone narrows, the intact zone above the failed zone provides higher strength due to confinement. This condition either limits or entirely prevents further failure or considerably delays the process and eventually self-arrests the degradation process (i.e., no rockfall). Both DOE's results and the NRC staff's independent calculations show that the thermal degradation should stop within one radius of depth into the drift's roof. On the basis of the NRC staff's confirmatory calculation, DOE's assertion is reasonable that under the repository mechanical and thermal stress conditions, the confined rock strength at one radius depth is more than twice the unconfined compressive strength. Therefore, for rock Categories 2 to 5, thermal degradation should be limited to depths shallower than one radius above the drift's crown.

Model Support and Consistency With Available Observations

DOE supported the use of the UDEC–Voronoi model in the thermal-mechanical analyses through four investigations, as identified in BSC Section 7.6.5 (2004al). DOE compared modeled failure mechanisms to large-core lithophysal sample failure mechanisms observed in the laboratory. DOE concluded that the UDEC model could simulate the observed patterns of fracturing due to (i) the axial splitting failure mode of lithophysal samples in unconfined compression tests and (ii) the measured strength and Young's modulus of the samples. Modeled drift-scale fracturing of the lower lithophysal tuff in the Enhanced Characterization of the Repository Block Cross-Drift also compared favorably to observations of stress-induced tunnel sidewall fracturing in the Enhanced Characterization of the Repository Block Cross-Drift.

DOE conducted detailed modeling of the Drift-Scale Heater Test to determine whether the UDEC model could reasonably represent the spallation of nonlithophysal tuff observed during the test. Small amounts of spallation from the drift crown were observed during the heater tests. Once the UDEC model was calibrated to appropriate Topopah Spring nonlithophysal tuff characteristics, the model was able to calculate small amounts of rockfall from the overstressed crown of the heated drift. DOE provided additional details of this analysis in DOE Enclosure 7 (2009ae), including quantification and favorable comparison of the calculated and observed amounts of rockfall for this test.

DOE used a continuum-based approach to model elastic and inelastic rock stress for a range of conditions representative of heated drifts, as described in BSC Section 7.6.5.4 (2004al). Although DOE does not consider continuum-based models as appropriate for calculating rockfall due to thermal-mechanical processes, as identified in BSC Section 7.4.1 (2004al), DOE concluded that both the continuum and the discontinuum (UDEC) models appropriately represent stress distributions prior to reaching the yielding point of the rock.

DOE supported the use of the calibrated rock-mass characteristics by comparing laboratory experiments of lithophysal rocks, as detailed in BSC Section 7.6.4 (2004al). DOE stated in BSC p. 7-61 (2004al) that the number and types of laboratory and *in-situ* experiments were insufficient to describe the complete constitutive behavior of the lithophysal tuff with a high level of confidence, particularly in the postpeak strain range and for confined conditions. Consistent with common engineering practice, DOE analyzed the continuum constitutive Mohr-Coulomb models ranging from perfectly plastic to perfectly brittle to bound the possible behavior of the lithophysal rock mass on damage and deformation. To accommodate the uncertainty represented by the limited characterization of the lithophysal tuff, DOE calibrated the UDEC–Voronoi model to give a more brittle stress-strain response than observed in tested samples, as described in BSC p. 7-38 (2004al). According to DOE, this approach enhanced the ability of rockfall to occur in the UDEC–Voronoi model, as identified in DOE Enclosure 6 (2009ae).

NRC Staff's Review of Model Support and Consistency With Available Observations

The NRC staff reviewed the information DOE provided to support its use of the UDEC model in the thermal-mechanical analyses for drift stability. A key element of the UDEC–Voronoi model is the representation of postpeak strain. DOE presented several analyses showing calculated postpeak strains for simulated rock masses in BSC Section 7.6.4 (2004al). DOE presented limited information on postpeak strain characteristics for the Topopah Spring lithophysal tuff. Although the single comparison between the lithophysal tuff and UDEC calculation for stress-strain characteristics showed a calculated response that is more brittle than exhibited by the laboratory experiment in DOE Enclosure 1 (2009cd), this information did not address the range of characteristics represented by the five lithophysal rock-mass categories used in the UDEC analyses. Additionally, strength characteristics for only six samples from the Topopah Spring lower lithophysal tuff are reported in BSC Table 6-69 (2007be). DOE Enclosure 1 (2009ce) provided additional information on the adequacy of the six samples to represent the range of strength characteristics needed to support the UDEC analyses in BSC Figure 7-16 (2004al).

In the response to a request for additional information related to the rock-mass categories [DOE Enclosure 1 (2009ce)], DOE stated that in modeling the rock mass responses, it applied a bounding approach to those five rock mass categories (lower bound relations between stiffness and strength cover and bound the loading response). This approach is meant to encompass the variability and uncertainties of the laboratory and field data. For postpeak response, the UDEC–Voronoi block model was calibrated to bound the brittleness of the lithophysal rock mass observed from the experimental data. This was achieved by bounding all test data in the axial stress versus axial strain curve, as outlined in DOE Enclosure 1 (2009ce). This approach is reasonable, because biasing the model calibration to a more brittle response than observed in real rock will not underestimate the potential for rockfall to occur.

Treatment of Time-Dependent Failure

Time-dependent failure refers to the potential for rock to fail by gradual weakening under stresses less than the rock strength, if the rock is subjected to that stress for long periods of time. DOE considers the potential for time-dependent failure as a function of the ratio of applied stress to the rock strength. DOE evaluated the relationship of time to failure on the basis of two sets of test data for stress ratios ranging from about 0.8 to 1.0. A best linear fit between the stress ratio and the logarithm of time was calculated and used to extrapolate times to failure for stress ratios less than 0.8. For the extrapolated portion of this curve, predicted times to failure ranged from approximately 12 days (10^6 s) at a ratio of 0.8 to about 32,000 years (10^{12} s) at a ratio of 0.6. Below values of 0.55, no time-dependent failure is predicted. In BSC Appendix S (2004al), DOE supported the use of a linear fit approximation by comparison to a previous study of data from Lac du Bonnet granite and concluded that the linear fit is appropriate. DOE evaluated the uncertainty in the time-to-failure estimates by running the UDEC model for rock Categories 1, 2, and 5 using times to failure based on the Lac du Bonnet data.

In the response to the NRC staff's request for additional information related to the linear relationship fit to represent the time-to-failure versus stress ratio data for tuff [DOE Enclosure 3 Number 2 (2009ae)], DOE acknowledged uncertainty in the data used for the linear fit and cited observations from the Enhanced Characterization of the Repository Block and Exploratory Studies Facility as additional evidence that time to failure is not overestimated. DOE stated that stress ratios in the range of 0.58 to 1.0 are represented at unsupported drift spring-lines for a longer time (greater than 10 years) than is available from any experiment, and no significant degradation has occurred.

NRC Staff's Review of Treatment of Time-Dependent Failure

The NRC staff evaluated the extent to which time-to-failure estimates could affect predicted drift degradation, especially in the range of stress ratios between 0.6 and 0.7, for which time-to-failure data for tuff are not available but relatively long times to failure are predicted (i.e., 32 years for a ratio of 0.7 and to 32,000 years for a ratio of 0.6). There is uncertainty in these estimates because the data points are few and the correlation coefficient for the linear fit to the data is relatively low as shown in BSC Figure S-27 (2004al). Numerical analyses by DOE, shown in BSC Figures S-14 through S-21 (2004al), also suggested times to failure for this range of stress ratio could be on the order of a few days to a few years. In DOE Enclosure 4 (2009ae), DOE cited observations from the Enhanced Characterization of the Repository Block and Exploratory Studies Facility tunnels, stating that these tunnels represent stress ratios between 0.58 and 1.0; however, significant rock failure has yet to occur.

In DOE Enclosure 1 (2009cg), DOE indicated that the uncertainty in time-dependent strength degradation of the lithophysal tuff was not represented in the thermo-mechanical calculations of drift stability, because the static-fatigue curve, based on the 1997 tuff data, bounds the potential for thermally induced drift degradation. Bounding was achieved by applying the Lac du Bonnet static-fatigue relationships for granite to the lithophysal tuff data. The NRC staff notes that this approach underestimates the time to failure for tuff and, thus, the analytic model maximizes the potential for thermally induced drift degradation by calculating degradation earlier than expected, as shown in BSC Figure S-30 (2004al). DOE also indicated that temperatures in the range between ambient and 200 °C [392 °F] have a small effect on the tuff mechanical properties, including short-term strength and time to failure. The NRC staff reviewed DOE's information and notes this relationship is reasonable. The static-fatigue curve for tuff, based on the 1997 and 2004 DOE data sets, predicts more rapid drift degradation than the observed conditions in the Exploratory Studies Facility and Enhanced Characterization of the Repository Block Cross-Drift. The DOE approach for modeling time-dependent failure is reasonable, because this bounding approach will not underestimate the amount of rockfall for lithophysal rocks.

Alternative Conceptual Models

DOE considered alterative conceptual models that were based on assumptions and simplifications that differed from those of the base-case models discussed previously and described in BSC Section 6.7 (2004al). The conceptual models DOE considered included continuum models. In a continuum model, the lithophysae and fractures are smeared into the elements of a continuous rock mass, where there is no slip between model elements. In the discontinuum model, lithophysae and fractures are represented by joints between the Voronoi blocks and slip can occur between these model elements. Although a continuum model can simulate the accumulation and distribution of stress prior to yielding, the model cannot accurately represent stress-strain relationships once the unconfined compressive strength of the rock is reached. Thus, DOE concluded a continuum-based approach is inappropriate for representing rockfall in lithophysal rock, because the relatively ductile characteristics of this rock type require an understanding of post-peak stress response. Nevertheless, DOE did use a continuum model to evaluate the thermal-mechanical conditions for the discontinuum model, prior to initiation of rockfall.

NRC Staff's Review of Alternative Conceptual Models

The NRC staff reviewed the technical basis DOE provided in its evaluation of alternative conceptual models to the UDEC–Voronoi approach. As discussed in previous sections, DOE characterized the stress-strain relationships expected for lithophysal tuff. This characterization showed that the lithophysal tuff is not expected to fail once the unconfined compressive strength is reached and that postpeak strength is available through ductile deformation to accommodate additional stress. In contrast, a continuum-based approach assumes that there is no post-peak strength to the strained rock mass and that rock failure occurs once the unconfined compressive strength is reached.

The Center for Nuclear Waste Regulatory Analyses (CNWRA®) performed an independent analysis that considered a continuum-based model to assess the technical issues for thermally induced drift degradation. The drift degradation study (Ofoegbu, et al., 2007aa) used the Abaqus two-dimensional continuum method. That method, as applied by CNWRA, did not incorporate the physics of time-dependent stress-strain releases that occurs during the rock spalling process. The analysis did not account for the development of fracture structures leading to failure regions with variable extent of rock cohesion. Consequently, the results are unrealistically conservative. Nonetheless, CNWRA analyses led to the development of technical questions transmitted to DOE, as referenced in this section.

The NRC staff notes that a discontinuum-based approach, such as that used by the UDEC–Voronoi model, provides a more accurate representation of rock response to thermal-mechanical effects than a continuum-based approach. Although both NRC and DOE have used continuum-based models to evaluate stress distributions around heated drifts and to provide insights on rock mechanical processes, the NRC staff notes that continuum-based models are not appropriate for representing the stress-strain relationships that control the occurrence of rockfall in lithophysal tuff. DOE has considered continuum-based models and has provided a reasonable basis to exclude the use of these models, per BSC Section 6.4.2 (BSC, 2004al), in the performance assessment.

Summary of DOE's Analysis and NRC Staff's Review

DOE adequately analyzed the thermally induced stresses causing instability of the waste emplacement drifts, compared the calculated stresses to the estimated strength of the rock mass, and estimated the timing and extent of potential drift degradation under anticipated loads. The methodology is reasonable because such analyses allow a systematic study of potential rock mass behavior under a range of anticipated loading scenarios. The NRC staff has reviewed SNL (2008ab), associated references (BSC, 2004al), and responses to the NRC staff's requests for additional information in its evaluation of DOE's exclusion of drift collapse due to thermal stresses and time-dependent rock weakening. The NRC staff performed independent confirmatory analyses in its evaluation of DOE's application. The NRC staff's independent analyses (Cao, 2010aa), using standard analytic techniques, confirmed DOE's conclusions.

DOE accounted for variability in rock types and a range of mechanical properties and strength characteristics, on the basis of laboratory tests and field investigations. DOE has presented technical bases for its conclusions that rockfall in lithophysal rocks, with natural fractures and weak planes along which preferential failures occur, can be evaluated by the discontinuum models. DOE has used reasonable technical approaches for quantifying the amount of rockfall that potentially results from nonseismic-induced drift collapse. DOE's methods to quantify the amount of thermally induced rockfall are reasonable. Thus, DOE's conclusion that combined effects of mechanical, thermal, and time-dependent weakening of rocks can be excluded from its performance assessment is reasonably supported.

Summary of NRC Staff's Review for FEP 2.1.07.02.0A, Drift Collapse

The NRC staff reviewed the models and results DOE used for screening out thermally induced drift degradation at the proposed Yucca Mountain repository using risk-informed, performance-based review methods described in the YMRP. Significant aspects of this review included determining whether DOE used reasonable model domains and boundary conditions, appropriate initial stress states and temperature inputs, appropriate rock block characteristics in the UDEC model, and suitable methods to calibrate and support the UDEC model. On the basis of the results of this review, the DOE technical basis for excluding FEP 2.1.07.02.0A, Drift Collapse, is reasonable.

FEP 2.1.07.05.0B, Creep of Metallic Materials in the Drip Shield

DOE excluded Creep of Metallic Materials in the Drip Shield from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 5 (2009af). Creep refers to time and temperature-dependent plastic (i.e., permanent) deformation of material caused by static loading. DOE used the feature, event, and process Creep of Metallic Materials in the Drip Shield to consider creep as a potential degradation process affecting the drip shield. Due to the long 10,000-year period of consideration and the possibility of early drift collapse after the waste emplacement, DOE noted the importance of the analysis of time-dependent deformation and the stability of the drip shield when nonuniformly loaded by the rock rubble mass.

DOE developed constitutive equations to express the amount of creep strain for Titanium Grades 7 and 29 as a function of temperature, applied stress, and time. DOE Enclosure 5, Section 1.2.1 (2009af) assumed a drip shield temperature of 150 °C [302 °F] for the screening analysis. DOE stated that higher drip shield temperatures would only be reached in the event of near-complete drift collapse within the first few hundred years after repository closure and that, even for early drift collapse, the temperature will drop below 150 °C [302 °F] within 600 to 1,000 years after waste disposal. DOE concluded that it was reasonable to assume a constant temperature of 150 °C [302 °F] for the screening analysis because the creep susceptibility of titanium alloys generally decreases with decreasing temperature and 150 °C [302 °F] is an overestimate of the drip shield temperature for most of the postclosure period.

In BSC Attachment I (2005an), DOE used titanium creep data from the literature to derive creep equations for Titanium Grades 7 and 29 at 150 °C [302 °F]. Because there are limited creep data in the literature for titanium alloys for temperatures around 150 °C [302 °F], DOE first derived equations to represent the creep behavior at room temperature, then rescaled those equations to a temperature of 150 °C [302 °F] using information about effects of temperature on creep kinetics. To derive the room temperature creep equation for Titanium Grade 7, DOE fitted a power-law-type equation to the 27-year creep data for Titanium Grade 2 from Drefahl, et al. (1985aa). DOE used BSC Eq. (I-8) (2005an) to represent the room-temperature creep of Titanium Grade 7. To derive the room-temperature creep equation for Titanium Grade 29, DOE fitted a power-law-type equation to the 1,000-hour creep data for Titanium Grade 5 from Odegard and Thompson (1974aa). DOE used BSC Eq. (I-19) (2005an) to represent the room-temperature creep of Titanium Grade 29. To rescale the room-temperature creep equations to represent the creep behavior at 150 °C [302 °F], DOE first accounted for the difference in yield stress at the respective temperatures, using BSC Eq. (I-7) (2005an). DOE then rescaled the creep equations using BSC Eq. (I-12) (2005an) assuming an activation energy of 30 kJ/mol. In this manner, DOE derived BSC Eqs. (I-15) and (I-22) (2005an) to represent the creep behavior of Titanium Grades 7 and 29, respectively, at 150 °C [302 °F]. DOE compared the creep strains the equations calculated to literature data for creep of titanium alloys at 150 °C [302 °F] (Kiessel and Sinnott, 1953aa; Odegard and Thompson, 1974aa). DOE stated that the equations used to represent the creep behavior of Titanium Grades 7 and 29 at 150 °C [302 °F] are reasonable because they predict greater creep strain than reported in the technical literature.

In the second part of the DOE creep analysis, DOE performed a finite element structural analysis of the drip shield, considering six potential loading scenarios derived from BSC (2004al) and using the constitutive creep equations to analyze the extent of drip shield creep. DOE assumed that creep will cause the drip shield to collapse when tertiary creep begins at any point on the drip shield. Tertiary creep refers to a rapid increase in creep strain rate associated with material instability, leading to rupture. DOE assumed a tertiary creep threshold of 10 percent strain and concluded that this threshold is conservative because experimental observations (Drefahl, et al., 1985aa) indicated that the onset of tertiary creep in titanium alloys occurs at about 15 percent strain. On the basis of creep analyses cited in the feature, event, and process screening justification (SNL, 2008ab), DOE concluded that the maximum strain is below the onset strain for tertiary creep. Therefore, DOE concluded that creep would not impact the drip shield's ability to divert seepage and protect the waste package from anticipated loads. DOE concluded that it is appropriate to exclude the feature, event, and process Creep of Metallic Materials in the Drip Shield from the performance assessment model on the basis of low consequence (SNL, 2008ab).

NRC Staff's Review

The NRC staff reviewed DOE's justification for assuming a constant temperature of 150 °C [302 °F] for the creep analysis. In BSC (2005an), DOE represented titanium creep as a thermally activated process, where the susceptibility to creep increases with increasing temperature. The treatment of creep as a thermally activated process is consistent with the technical literature (Orava, 1967aa; Stetina, 1969aa; Zeyfang, et al., 1971aa; Miller, et al., 1987aa). In BSC Assumption 3.2.4 (BSC, 2005an), however, DOE stated that the drip shield temperature may exceed 150 °C [302 °F] for several hundred years in the event of early drift collapse. This suggests that, in the event of early drift collapse, the susceptibility of the drip shield to creep could be greater than represented by the DOE analysis for 150 °C [302 °F]. As such, the NRC staff submitted an request for additional information to DOE requesting that it provide the rationale for using 150 °C [302 °F] as the analysis temperature. DOE stated in DOE Enclosure 5, Section 1.2.1 (2009af) that 300 °C [572 °F] is a reasonably bounding temperature because there is less than 10^{-4} probability that a drip shield will exceed this temperature for early drift collapse. Further, in DOE Enclosure 5, Section 1.2.3 (2009af), DOE stated that the creep equations for 150 °C [302 °F], will not underestimate the extent of creep at 300 °C [572 °F] because above 150 °C [302 °F] creep becomes an athermal process (i.e., the susceptibility to creep does not increase with temperature). DOE attributed this behavior to the phenomenon of dynamic strain aging: a process whereby solute impurity atoms diffuse to areas of dislocations and impede dislocation motion. The NRC staff reviewed the technical literature and confirmed that investigators (Moskalenko and Puptsova, 1972aa; Stetina, 1980aa) have reported a transition in creep control from thermal to athermal processes. There is some uncertainty in the transition temperature, as values were reported in the range of 150 to 400 °C [302 to 752 °F]. The NRC staff recognizes on the basis of the cited references, however, that the transition temperature tends to decrease with decreasing strain rate and approaches 150 °C [302 °F] for the low strain rates generally associated with creep. Therefore, DOE's representation of creep as an athermal process at temperatures above 150 °C [302 °F] is reasonable. DOE's assumption that the drip shield temperature is 150 °C [302 °F] for the creep analysis is reasonable because (i) creep is likely independent of the temperature at temperatures above 150 °C [302 °F] and (ii) the drip shield could experience temperatures above 150 °C [302 °F] only during a relatively short period compared to the 10,000-year period considered in the creep analysis. Thus, because of (i) and (ii), DOE did not underestimate the amount of creep strain in its analysis for the postclosure period. The evaluation of DOE's temperature computation is addressed in TER Section 2.2.1.3.6, where the NRC staff noted that temperature computations were appropriate for their intended use within the performance assessment model.

The NRC staff reviewed DOE's methodology to develop equations to represent the creep of Titanium Grades 7 and 29 at room temperature. With respect to Titanium Grade 7, DOE's approach to consider published empirical creep data (Drefahl, et al., 1985aa) as input to the analysis is reasonable. The NRC staff notes that the difference in chemical composition between Titanium Grades 2 and 7 is the addition of a small amount of palladium in the latter, which has a minimal effect on creep behavior because it does not significantly change the alloy microstructure. Moreover, the Titanium Grade 2 material Drefahl, et al. (1985aa) studied had large grain sizes, which, according to the technical literature (e.g., Ankem, et al., 1994aa; Aiyanger, et al, 2005aa) makes it susceptible to creep at temperatures from room temperature to 150 °C [302 °F]. On the basis of this information, DOE's use of the creep data from Drefahl, et al. Figure 3 (1985aa) to model the creep behavior of Titanium Grade 7 in the drip shield is reasonable. Because BSC Eq. (I-8) (2005an) calculates greater creep strain than Drefahl, et al., Figure 3 (1985aa), the use of this equation to represent the room-temperature creep of Titanium Grade 7 is reasonable.

DOE's approach to consider published empirical creep data for Titanium Grade 5 (Odegard and Thompson, 1974aa) as input to the creep analysis of Titanium Grade 29 is reasonable. The NRC staff notes that the difference in chemical composition between Titanium Grades 5 and 29 is the addition of a small amount of ruthenium in the latter, which is expected to have a minimal effect on the creep behavior because it does not significantly change the alloy microstructure. Odegard and Thompson (1974aa) studied thermally aged Titanium Grade 5; DOE described that the microstructure of Titanium Grade 5 is similar to Titanium Grade 29 given the small differences in composition between Grades 5 and 29. On the basis of this information, DOE's use of the creep data from Odegard and Thompson Figure 3 (1974aa) to model the creep behavior of Titanium Grade 29 in the drip shield is reasonable. Because BSC Eq. (I-19) (2005an) calculates greater creep strain than Odegard and Thompson Figure 3 (1974aa), the use of this equation to represent the room-temperature creep of Titanium Grade 29 is reasonable.

The NRC staff reviewed DOE's methodology to rescale the room-temperature creep equations for Titanium Grades 7 and 29 to 150 °C [302 °F]. In rescaling the room-temperature creep equations, DOE accounted for the temperature effect twice: once using the difference in yield stress for the respective temperatures and again using the activation energy. DOE asserted that this redundancy is conservative because the activation energy alone should quantify the effects of temperature on creep kinetics. The NRC staff noted, however, that there is uncertainty in the value of the activation energy for creep of titanium alloys. DOE's selected activation energy of 30 kJ/mol is lower than the activation energy of approximately 150 kJ/mol Kiessel and Sinnott (1953aa) and Stetina (1969aa) reported. In BSC (2005an), DOE represented the creep strain temperature-dependence as an exponential function of the activation energy, such that a small change in the activation energy would yield a large change in the calculated creep strain. Therefore, the NRC staff sent a request for additional information that DOE address how its methodology for rescaling the room-temperature creep equations to 150 °C [302 °F] accounts for the uncertainty in the creep temperature dependence. In DOE Enclosure 6 (2009af), DOE stated that the activation energy for titanium creep depends on the rate-limiting deformation mechanism which, in turn, depends on a number of parameters including the alloy microstructure, phase composition, and strain rate. DOE further stated that literature reports that give higher activation energy than used in its creep analysis do not provide sufficient information about the material and test conditions to support a direct comparison of the activation energies. DOE asserted, however, that conservative aspects of its approach to quantify creep temperature dependence yield creep equations that calculate greater creep strains than have been experimentally measured for Titanium Grades 7 and 29 in the temperature range of room temperature to 150 °C [302 °F].

The NRC staff reviewed the information DOE provided in DOE Enclosure 6 (2009af). The NRC staff compared the creep strains DOE's temperature-scaled creep equations calculated to literature values of creep strain at temperatures comparable to 150 °C [302 °F]. The NRC staff confirmed that for Titanium Grade 7, the DOE calculated greater creep strain at 125 °C [257 °F] than Teper (1991aa) measured for Titanium Grade 2 at that temperature. Further, DOE calculated greater creep strain at 99 and 204 °C [210 and 399.2 °F] than Kiessel and Sinnott (1953aa) measured for commercially pure titanium at these temperatures. For Titanium Grade 29, the NRC staff confirmed that DOE calculated greater creep strain at 66 and 149 °C [150 and 300.2 °F] than Thompson and Odegard (1973aa) measured for Ti-5Al-2.5Sn at these temperatures, even though Ti-5Al-2.5Sn has greater susceptibility to creep than Titanium Grade 29. In spite of uncertainty in the creep activation energy, DOE overestimated the creep strain, in part, because it used creep data from alloys that had microstructures particularly susceptible to creep for deriving the room-temperature creep equation. Moreover, DOE accounted for the temperature dependence of creep using the difference in yield stress at room temperature and 150 °C [302 °F], in addition to the activation energy, whereas the effects of temperature on creep kinetics should be physically quantified only in the latter. On the basis of this information, DOE's use of BSC Eqs. (I-15) and (I-22) (2005an) to represent the creep behavior of Titanium Grade 7 and 29, respectively, at 150 °C [302 °F] is reasonable, because these equations do not underestimate the creep strain of the drip shield.

The NRC staff reviewed DOE's assumption that a creep strain of 10 percent anywhere on the drip shield will cause its collapse and that any strain smaller than that will not significantly affect the drip shield. Long-term creep data for Titanium Grades 2 and 5 from Drefahl, et al. (1985aa) show a transition from steady-state secondary creep to unstable tertiary creep at creep strain of approximately 15 percent. The NRC staff expects that the creep behavior of Titanium Grades 7 and 29 will be analogous to those of Titanium Grades 2 and 5, respectively, because the addition of a small amount of palladium or ruthenium will not significantly affect the alloy microstructure. Therefore, 10 percent strain is a reasonable threshold for the onset of tertiary creep because it does not underestimate the threshold strain.

The NRC staff reviewed DOE's finite difference structural analyses on creep deformation of the drip shield exposed to six loading scenarios (BSC, 2004al) presented in BSC (2005an). In these analyses, DOE considered the highest vertical pressure applied to the drip shield crown of 154.81 kPa [22.45 psi]. The NRC staff notes the range of loads DOE considered is reasonable. This is further addressed in TER Section 2.2.1.3.2.

In summary, (i) DOE did not underestimate the amount of creep strain in its analysis for the postclosure period, (ii) DOE developed reasonable equations to represent the creep of Titanium Grades 7 and 29 at room temperature, (ii) DOE's methodology to rescale the room-temperature creep equations for Titanium Grades 7 and 29 to 150 °C [302 °F] is reasonable, (iv) DOE's use of 10 percent strain is a reasonable threshold for the onset of tertiary creep because it does not underestimate the threshold strain, and (v) DOE considered a reasonable range of loads. On the basis of the results of this review, the DOE technical basis for excluding FEP 2.1.07.05.0B, Creep of Metallic Materials in the Drip Shield, is reasonable.

FEP 2.1.09.03.0B, Volume Increase of Corrosion Products Impacts Waste Package

DOE excluded Volume Increase of Corrosion Products Impacts Waste Package from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosures 10–11 (2009ab). In the feature, event, and process, DOE considered volume increase of corrosion products (increase due to the higher molar volume of corrosion products than intact, uncorroded material) from the waste form, cladding, and waste package as a mechanism that could damage the waste package.

DOE excluded the effect of volume increase of corrosion products on the basis of the following: (i) if the outer container is not breached, there will be negligible corrosion products; (ii) there are unlikely events leading to early waste package outer container failure; (iii) extended time (thousands of years) is needed for corrosion products to fill the space between the outer and inner containers before any significant stress buildup occurs; (iv) due to the higher Alloy 22 mechanical strength compared to the stainless steel strength, there is a higher likelihood for the inner stainless steel container to deform or crack if additional stresses develop due to the corrosion product buildup; and (v) extensive time is needed for the development of stresses needed to promote stress corrosion cracking on the waste package outer container.

The NRC staff reviewed the summary technical basis in the FEP document (SNL, 2008ab) and DOE Enclosures 10–11 (2009ab). DOE stated that prior to breach of the Alloy 22 waste package outer container, only dry oxidation by residual moisture is possible on the Alloy 22 inner surface or on the surface of the stainless steel inner container. DOE concluded that the residual moisture in the waste package will not result in a large volume of corrosion products to cause mechanical damage to the Alloy 22 or stainless steel container. DOE's conclusion is adequate because waste packages are expected to include minimal residual moisture that is not sufficient to significantly oxidize metallic containers.

DOE assessed that for significant corrosion of Alloy 22 inner surface and the stainless steel container, the Alloy 22 outer container must first be breached. The Alloy 22 general corrosion rates are low. Stress corrosion cracking in the absence of weld flaws or seismic activity would not breach the outer container in 10,000 years after waste emplacement, according to DOE. Combinations of large flaws and stresses are uncommon, and large magnitude seismic events capable to fail the waste packages within the first 10,000 years after waste emplacement are rare, according to DOE. Nonetheless, to address the case of container failure due, for example, to seismic events, DOE assumed failure of the outer container and conducted two analyses to estimate the magnitude and timing of stresses on the waste package outer container from the corrosion products from the inner vessel corrosion. DOE performed analyses, as detailed in DOE Enclosure 10 (2009ab), to show that stresses sufficient to enhance degradation of the outer container would not develop within 10,000 years after breach of the waste package outer container.

In DOE's assessment of the dependence of volume increase of corrosion products on outer container corrosion, DOE considered information on Alloy 22 general corrosion rates. The NRC staff noted that these corrosion rates were consistent with DOE's general corrosion model evaluated in TER Section 2.2.1.3.1.3.2. On the basis of the NRC staff's determinations in TER Section 2.2.1.3.1.3.2 that the laboratory test results and models for long-term prediction were reasonable, the Alloy 22 corrosion rates were adequate for their intended use within the performance assessment model. Thus, the assessment of the effect of outer container corrosion on the volume increase of corrosion products was reasonable. With regard to early failure, localized corrosion, or igneous intrusion model cases, DOE stated in DOE Enclosure 10 (2009ab) that the performance assessment for these model cases does not take credit for the further presence of the waste package; thus, volume increase of corrosion products would not change the estimated consequences.

On the basis of DOE Enclosures 10–11 (2009ab), the NRC staff notes that DOE assessed all possible corrosion modes for the inner and failed outer containers. The corrosion modes that DOE considered include crevice corrosion, stress corrosion cracking, and galvanic corrosion. DOE estimated that the gap between the inner and outer containers would be filled with corrosion products after thousands of years (between 1,400 and 37,000 years) after breaching of the outer container (SNL, 2008ab). The time for stress buildup sufficient to cause stress corrosion cracking on the waste package outer container would exceed 10,000 years after the initial waste package breach, according to DOE Enclosure 10 (2009ab). The NRC staff notes that uncertainties remain with respect to the magnitude of stainless steel corrosion rates, their environmental dependence, and longer term values (He, et al., 2007ab). Higher stainless steel corrosion rates could fill the gap with corrosion products and cause stress buildup earlier than estimated in DOE's analyses, increasing the waste package cracked area. However, DOE described that the extent of the area compromised by cracks is overestimated by the consideration of a crack distribution that fills a two-dimensional space, and consideration of a stress level equal to the yield strength of the material (as opposed to allowing the stress to relax when cracks form or grow), as detailed in DOE Enclosures 10 and 5, respectively, (2009ab,cj) and SNL Section 6.7.3 (2007bb). On the basis of DOE's information, and supported by the NRC staff's evaluation of stress corrosion crack size and density in TER Section 2.2.1.3.1.3.2.3 (wherein the NRC staff noted that DOE appropriately abstracted the dimension of waste package area damaged by stress corrosion cracking for the intent of the performance assessment), the NRC staff notes that DOE provided adequate support for its conclusion that additional stresses from the stainless steel corrosion products are unlikely to significantly increase the extent of the waste package area covered by cracks. In the case of large weld flaws leading to stress corrosion cracking initiation, DOE's conclusion that the results of the performance assessment would not significantly change is reasonable; DOE reached this conclusion by considering stress buildup from stainless steel corrosion products leading to a larger waste package area covered by cracks (larger than the weld cracks alone). This is because DOE stated (i) large welds flaws leading to stress corrosion cracking are rare and (ii) it could take thousands of years for enough corrosion product buildup to fill inner and outer container gaps and even longer to develop sufficient stress buildup. Therefore, on the basis of the DOE analyses considering a complete range of waste package failure modes, computation of the time to fill gaps and produce significant stresses on the waste package outer container, and computations of the area compromised by cracks that are likely to overestimate the waste package damage area, the exclusion of the feature, event, and process Volume Increase of Corrosion Products Impacts Waste Package by low consequence is reasonable.

FEP 2.1.09.28.0A, Localized Corrosion on Waste Package Outer Surface Due to Deliquescence

DOE excluded Localized Corrosion on Waste Package Outer Surface Due to Deliquescence on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosures 12–15 (DOE, 2009ab). In the feature, event, and process, DOE considered that moisture from air could be absorbed by salts in dust deposited on the waste package, even at low relative humidity; this moisture could dissolve the salts and create concentrated aqueous solutions or brine. According to DOE, these brines could promote localized corrosion of the waste package outer surface.

DOE's analysis of the penetration of the Alloy 22 waste package outer barrier by localized corrosion induced by dust deliquescence brines was based on the following five questions from SNL, pp. 6-705 to 6-710 (2008ab):

1. Can multiple-salt deliquescent brines form at elevated temperature?
2. If deliquescent brines form at elevated temperature, will they persist?
3. If deliquescent brines persist, will they be corrosive?
4. If deliquescent brines are potentially corrosive, will they initiate localized corrosion?
5. Once initiated, will localized corrosion penetrate the waste package?

In SNL (2008ab), DOE stated that the answers to those questions are (1) yes, (2) sometimes, (3) not expected, (4) no, and (5) no, respectively. Because all of the questions must be answered affirmatively for outer container penetration to be possible, DOE concluded that localized corrosion was unlikely. In summary, DOE concluded that brines formed by dust deliquescence are not expected to be aggressive; the amount of brine volume that will be distributed on the waste package will be extremely small and will not support the initiation of localized corrosion; and several processes will stifle localized corrosion limiting, penetration of the waste package outer container (SNL, 2008ab). Accordingly, DOE excluded Localized Corrosion on Waste Package Outer Surface Due to Deliquescence from the performance assessment model.

NRC Staff's Review

The NRC staff reviewed the technical basis in the feature, event, and process document (SNL, 2008ab), additional information in BSC (2005aa) and SNL (2007al), and the analysis to supplement the screening justification in DOE Enclosures 12–15 (2009ab). DOE provided a key technical basis: the brine volume will be extremely small {2 $\mu L/cm^2$ [7.87×10^{-4} in^3/in^2]} and it will be mixed with a large amount of insoluble dust on the waste package surfaces in the repository setting. Under this condition, the NRC staff notes that DOE's conclusion in DOE Enclosures 12–15 (2009ab) that localized corrosion will not initiate nor propagate even if initiated is reasonable. DOE provided preliminary experimental results to support the analysis, obtained with specimens made of Alloy 22 and a series of less corrosion resistant analog materials (Inconel® 825, Hastelloy® C-276, and 80:20 Ni:Cr alloy). Some of the specimens were creviced specimens formed with a polytetrafluoroethylene-lined ceramic former and coated with a layer of salt mixtures expected to deliquesce under the repository conditions. DOE considered that the salt loading in the tests was greater than expected on the waste packages, as identified in DOE Enclosure 13, p. 6 (2009ab). The specimens were placed in a humidity chamber at 180 °C [356 °F] and at a relative humidity that enabled the coated salts to deliquesce. After an exposure of 25 or 50 days, the specimens were examined and no signs of localized corrosion were observed for Alloy 22, or in the less corrosion resistant Inconel 825, as described in BSC Section 6.4.2.2(a) (2005aa) and DOE Enclosures 12–15 (2009ab). On the basis of these short-term experiments showing that localized corrosion did not initiate under specific conditions enabling deliquescence of salts, the NRC staff notes that DOE provided a technical basis to exclude the feature, event, and process from the performance assessment model. DOE's conclusion that there is no evidence that localized corrosion could initiate and be sustained for extended periods in deliquescent solutions is reasonable. Thus, DOE's technical basis to exclude the feature, event, and process from the performance assessment is reasonable.

FEP 2.1.11.06.0A, Thermal Sensitization of Waste Packages

DOE excluded Thermal Sensitization of Waste Packages from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosures 16–17 (2009ab). According to DOE's definition of the feature, event, and process, phase changes in waste package materials could result from long-term storage under repository thermal conditions; phase changes could affect the corrosion resistance and mechanical properties of waste package materials. DOE described a model for long-term thermal aging and phase stability of Alloy 22 based on experimental measurements and theoretical calculations (BSC, 2004ab). The phase stability studies included tetrahedrally close-packed phase precipitation in the base metal and in the welded regions, and long-range ordering reactions. DOE conducted thermodynamic and kinetic modeling to predict the rate of precipitation of tetrahedrally close-packed phases and long-range ordering in Alloy 22 using the Thermo-Calc and DICTRA software and databases. DOE assessed validity of the aging and phase stability model and the databases in BSC (2004ab) and DOE Enclosures 16–17 (2009ab). According to the calculated time-temperature-transformation diagrams for the formation of P, σ, and ordered phases in Alloy 22 base metal, DOE stated that even if the temperature were to remain at the peak temperature for all time (which is an extremely conservative consideration), the transformation would still not have progressed 5 percent to completion after well over 1 million years. According to DOE, the planned solution annealing and quenching conditions for the waste package outer container are sufficient to prevent phase instability in Alloy 22. DOE compared the model results to the extent of tetrahedrally close-packed phase precipitation obtained from short-term aging experiments at temperature ranges exceeding those expected in the repository and the extent of long-range ordering from microhardness measurements. On the basis of these results, DOE concluded that insignificant aging and phase instability would occur in Alloy 22 under conditions that bound repository temperatures DOE estimated.

DOE also evaluated the effects of welding and thermal aging on the corrosion rate and localized corrosion resistance of Alloy 22 in the mill-annealed, as-welded, and as-welded plus thermally aged conditions (SNL, 2008ab, 2007al). On the basis of the results of short-term electrochemical tests, DOE stated that thermal aging and phase instability do not adversely affect the corrosion resistance of Alloy 22. In summary, DOE concluded that, on the basis of the model predictions and experimental evidence, long-term thermal aging is insignificant and phase instability is not expected to adversely affect the corrosion resistance of the waste package outer container. Therefore, DOE excluded the feature, event, and process from the performance assessment model on the basis of low consequence. As described in DOE Enclosure 16 (2009ab), DOE has imposed a restricted Alloy 22 composition range (SAR Table 1.9-9, Design Control Parameter 03-19), which has a narrower range of chemical compositions for Cr, Mo, Fe, and W compared to the composition limits specified in the standard ASTM B 575-04 (ASTM International, 2004aa). According to DOE, the design properties for Alloy 22 are in compliance with the ASME SB–575 specification (American Society of Mechanical Engineers, 2001aa).

NRC Staff's Review

The NRC staff reviewed DOE's technical basis for excluding the feature, event, and process and its model assumptions and model support in areas related to long-term thermal aging and phase stability of Alloy 22, as detailed in DOE Enclosures 16–17 (2009ab) and BSC (2004ab). The NRC staff has performed independent analyses of potential effects of thermal exposures to elevated temperatures on the phase stability of Alloy 22 and noted that thermal aging and fabrication processes could enhance precipitation of tetrahedrally close-packed phases (Pan, et al., 2005aa) and decrease the localized corrosion resistance of Alloy 22 (Dunn, et al., 2006aa). However, the potential effect of this decreased corrosion resistance is bounded by DOE's general and localized corrosion model abstractions for the waste package outer container, which as described by DOE, tend to overestimate the extent of general corrosion damage (due, for example, to a bias toward higher corrosion rates to account for differences in sizes of experimental metal coupons and size of "patches" on the waste package surface for the performance assessment computations, consideration of an enhancement factor to account for microbially enhanced corrosion, and other factors discussed in TER Section 2.2.1.3.1.3.2.1) and the frequency of localized corrosion (e.g., as discussed in TER Section 2.2.1.3.1.3.2.2, DOE's localized corrosion model predicts initiation of localized corrosion for some environmental conditions under which localized corrosion is not experimentally observed). The NRC staff reviewed DOE's description of the extent of general corrosion damage and the frequency of localized corrosion in TER Section 2.2.1.3.1.3.2 and, as described therein, notes that DOE's approach tends to overestimate the extent of general corrosion damage and the frequency of localized corrosion and is reasonable. The mechanical properties of the Alloy 22 fabrication welds will meet or exceed ASME SB–575 (American Society of Mechanical Engineers, 2001aa)—specified minimum mechanical property requirements, as DOE indicated in DOE Enclosures 16–17 (2009ab). On the basis of DOE's thermal aging and phase stability analyses and DOE's description that abstractions for general and localized corrosion of the waste package outer container are bounding, DOE's conclusion that long-term phase stability of Alloy 22 is of low consequence is reasonable. Therefore, it is reasonable to exclude the feature, event, and process Thermal Sensitization of Waste Packages by low consequence (with respect to corrosion resistance and mechanical properties of Alloy 22).

FEP 2.2.07.05.0A, Flow in the Unsaturated Zone (UZ) From Episodic Infiltration

DOE excluded Flow in the Unsaturated Zone from Episodic Infiltration from the performance assessment model on the basis of low consequence (SNL,2008ab), supplemented its technical basis for exclusion in DOE Enclosure 8 (2009cb) and DOE Enclosure 1 (2009cc), and provided supplemental material relevant to the technical basis for exclusion in DOE Enclosure 5 (2009bo). This feature, event, and process refers to the influence of episodic flow on radionuclide transport in the unsaturated zone; specifically, transient flow arising from episodic infiltration events. DOE stated that episodic flow through and below the repository horizon is expected to be strongly attenuated by the overlying Paintbrush Tuff nonwelded (PTn) hydrogeologic unit.

DOE stated that periods of high precipitation and water percolation are expected to occur during future rain storms, and also that the porous rock matrix in the PTn unit is expected to strongly attenuate episodic percolation fluxes. DOE described the PTn unit as ranging from approximately 21 m [70 ft] to over 120 m [400 ft] within the repository area. DOE stated that flow attenuation by the PTn is predicted to yield steady flow below the PTn in the unsaturated zone, except for volumetrically insignificant rapid flow through preferential pathways in the PTn. DOE asserted that transient flow below the PTn may occur in the southern part of Solitario Canyon because the PTn is completely offset by the Solitario Canyon Fault. However, episodic flow is not expected to significantly affect performance, because the emplacement drifts would be located away from Solitario Canyon in the affected area.

DOE based the assessment of episodic flow attenuation on two transient, one-dimensional simulations reported in SNL Section 6.9(a) (2007bf). DOE observed that the maximum flux below the PTn for these two simulations was around 17 mm/yr [0.67 in/yr], compared to the overall percolation flux uncertainty for the post-10,000-year period of 51 mm/yr [2 in/yr]. DOE supported its analysis by citing other studies considering one-, two-, and three-dimensional simulations, all using earlier estimates for PTn parameters. DOE stated that the other studies, in general, show similar damping of percolation flux by the PTn matrix.

DOE further supported the assessment of episodic flow attenuation in the PTn using results from (i) a water-release test within the PTn (in Alcove 4), (ii) line surveys of fracture minerals in tunnels below the PTn, (iii) inferred stagnation of a wetting pulse below the channel of Pagany Wash, and (iv) inferred long residence times in the PTn on the basis of C-14 observations from boreholes.

DOE considered Cl-36 observations from tunnels below the PTn, some of which have a radioisotope signature indicating that a portion of the *in-situ* waters infiltrated during or subsequent to the period of nuclear device testing from 1954 through 1970. DOE concluded that high observed concentrations of Cl-36 in some samples taken from the Exploratory Studies Facility tunnel possibly indicate relatively small amounts of fracture flow penetrating as fast pathways, either steady or transient, through fault zones between the ground surface and the repository elevation. DOE used flow and transport models to examine the Cl-36 observations, concluding that the quantity of water penetrating the PTn as a result of fast pathways is approximately 1 percent of total infiltration, and characterized this quantity as negligible with respect to repository performance.

DOE also considered tritium data from boreholes and tunnels below the PTn. DOE concluded that some observations of tritium below the PTn within the Exploratory Studies Facility and Enhanced Characterization of the Repository Block tunnels, and from five boreholes, also have a bomb-pulse or post-bomb-pulse radioisotope signature. DOE's analysis of the data led to the following conclusions: (i) all of the elevated tritium observations from the Exploratory Studies Facility are associated with faults, (ii) the elevated observations in the boreholes may be associated with lateral flow from faults, and (iii) most elevated tritium observations from the Enhanced Characterization of the Repository Block are not associated with faults but may be associated with fast and focused (but not necessarily episodic) flow pathways.

DOE concluded that the PTn will attenuate most episodic flow, resulting in approximately steady-state flow in the repository host rock and below, and the volume of flow that could lead to episodic flow in the repository host rock is small. Therefore, DOE excluded the feature, event, and process Flow in the Unsaturated Zone from Episodic Infiltration from the performance assessment model.

NRC Staff's Review

Because both the modeling results and field observations are consistent with DOE's description of the Flow in the Unsaturated Zone from Episodic Infiltration feature, event, and process, the information DOE provided supports the concept that the PTn matrix has a strong potential for dampening large pulses with matrix imbibition and episodic flow is most likely to be associated with prominent structural features (e.g., faults and intensely fractured zones). The information provided supports DOE's conclusion that bomb-pulse tritium observations in boreholes below the PTn are likely associated with lateral flow from faults and localized fast flow pathways that are not necessarily episodic flow. However, DOE did not show that episodic flow below the PTn is entirely precluded based on a relationship of the episodic flow to prominent structural features, because (i) 11 of the 22 tritium observations in the Enhanced Characterization of the Repository Block exhibit a modern signature despite being located more than 100 m [330 ft] from a mapped fault or intensely fractured zone and (ii) the travel times through the PTn that DOE concluded were necessary to explain these observations without invoking episodic flow are more than an order of magnitude faster than those obtained from the calibrated parameters.

The NRC staff notes that the information DOE provided supports the DOE conclusion that episodic flow-has a low consequence for the performance assessment. DOE considered increased seepage into emplacement drifts to be the largest performance consequence that would arise from episodic flow, but expects that any additional seepage would be small relative to the difference in percolation flux considered during calibration of infiltration uncertainty. In TER Section 2.2.1.3.6.3.2, the NRC staff considers episodic flow in the larger context of DOE's representation of the spatial and temporal variability of ambient percolation flux above and through the proposed repository horizon during performance assessment. In its review of information related to flow above the repository horizon (TER Section 2.2.1.3.6.3.2), the NRC staff noted that (i) systematic increases in seepage arising from episodic flow are small relative to the difference in percolation flux considered during calibration of infiltration uncertainty using DOE's assessment of fast pathways, (ii) increases in seepage are comparable using a conservative assessment of episodic pathways, and (iii) DOE showed that calculated maximum mean annual dose in the first 10,000 years is not substantially sensitive to systematic changes in seepage considered during calibration of infiltration uncertainty. Thus, DOE's technical basis to exclude the feature, event, and process Flow in the Unsaturated Zone from Episodic Infiltration from the performance assessments on the basis of low consequence is reasonable.

FEP 2.2.08.03.0A, Geochemical Interactions and Evolution in the Saturated Zone

DOE excluded Geochemical Interactions and Evolution in the Saturated Zone on the basis of low consequence, as outlined in SNL (2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 1 (2009ai). According to DOE's feature, event, and process definition, groundwater chemistry and other characteristics may change over time as a result of disposal system evolution or from mixing with other waters. Geochemical interactions may lead to dissolution and precipitation of minerals along the groundwater flow path, affecting groundwater flow, rock properties, and sorption of radionuclides (SNL, 2008ab).

In DOE Enclosure 1 (2009ai), DOE further examined natural groundwater geochemical variations in the immediate vicinity and downgradient from Yucca Mountain as a function of space and time. DOE stated in DOE Enclosure 1 (2009ai) that chemical compositions exhibit spatial variability that may be related to mixing of waters. (The NRC staff evaluates DOE's model abstractions of flow paths in the saturated zone in TER Section 2.2.1.3.8.) DOE stated that temporal changes in properties that may affect radionuclide sorption, such as pH, temperature, and major ion chemistry, are gradual and fall within the range of groundwater chemistries it considered in developing the transport parameter (sorption coefficients or K_d) values used in the saturated zone transport model of the performance assessment (SAR Section 2.3.9.3).

In its model abstraction for radionuclide transport through the saturated zone, DOE assumed oxidizing conditions along the flow paths through the tuff and alluvium. DOE stated in DOE Enclosure 1 (2009ai) that redox potential has a strong effect on the transport of redox-sensitive radionuclides. DOE also stated that other groundwater conditions such as reducing zones that may affect radionuclide sorption are localized in extent and unlikely to be changed at a larger scale for at least 10,000 years after repository closure. To support this statement, DOE reasoned that (i) there is sufficient pyrite in reducing hydrogeological units of the saturated zone to sustain those reducing conditions, (ii) the long residence time of water in the saturated zone causes its oxidation state to be largely determined by water–rock interactions, and (iii) no current mechanism is known to support the concept that reducing zones will become more extensive along the saturated zone path (SNL, 2008ab). For these reasons, DOE excluded Geochemical Interactions and Evolution in the Saturated Zone from the performance assessment model on the basis of low consequence (SNL, 2008ab).

DOE also presented performance assessment calculations that indicated the radionuclides that contribute the most to the calculated mean annual dose during the first 10,000 years after repository closure are nonsorbing, and that radionuclides whose sorption is most affected by changes in these geochemical parameters (Pu-239 and -240, Np-237, and Se-79) only constitute about 20 percent of the total mean annual dose during the first 10,000 years after repository closure. DOE also indicated in DOE Enclosure 1 (2009ai) that for the igneous intrusion modeling case, the release rates of plutonium and neptunium are only slightly sensitive to K_d values in volcanic rocks, but not sensitive to K_d values in the alluvium.

NRC Staff's Review

The NRC staff reviewed the model assumptions and field and laboratory data DOE used to support its screening justification, as identified in SNL (2008ab) and DOE Enclosure 1 (2009ai). To address uncertainty associated with natural variations in pH, temperature, and major ion chemistry, DOE considered a range in aqueous water chemistries in developing parameter distributions for the model abstraction of radionuclide transport through the saturated zone for the performance assessment (SAR Section 2.3.9.3). The NRC staff is aware of information that suggests temporal variations in key geochemical parameters that may influence potential sorption in the regional aquifers around Yucca Mountain (Perfect, et al., 1995aa; Turner and Pabalan, 1999aa). The NRC staff notes, however, that the likely changes in radionuclide sorption from evolving groundwater geochemistry are adequately captured by the range of K_d distributions considered in the performance assessment. Although DOE did not consider variability in redox conditions in developing the transport parameter distributions, the available information indicates that reducing conditions are limited to localized areas that do not appear to be widespread on a regional scale, as described in SNL Appendix F (2007ba). On the basis of the foregoing considerations and discussion of information DOE provided, the exclusion of this feature, event, and process on the basis of low consequence is reasonable.

FEP 2.2.08.03.0B, Geochemical Interactions and Evolution in the Unsaturated Zone

DOE excluded the feature, event, and process of Geochemical Interactions and Evolution in the Unsaturated Zone on the basis of low consequence following SNL (2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 18 (2009ab). According to DOE's description of the feature, event, and process, the geochemical environment of the unsaturated zone may evolve over time in response to thermal and chemical perturbations introduced by the repository system. Precipitation or dissolution of minerals or changes in groundwater chemistry may affect the flow and composition of seepage into drifts or the transport of radionuclides in the near-field environment (SNL, 2008ab). In the screening justification, DOE considered (i) how elevated temperatures would affect geochemical interactions between water and rock in the vicinity of the emplacement drifts and (ii) how changes in water chemistry due to reactions with repository construction materials would subsequently affect flow and transport properties in the unsaturated zone (SNL, 2008ab). DOE, in DOE Enclosure 18 (2009ab), also considered how geochemical interactions between waste package effluent and the solids and ambient waters might affect radionuclide transport in the crushed tuff invert and in the unsaturated rock beneath the repository drift.

DOE cited model analyses of geochemical interactions that estimated how drift seepage chemistry and near-field flow properties would be affected by changes in temperature, pH, redox potential, ionic strength and other compositional variables, time dependency, precipitation or dissolution, and resaturation times, as described in SNL (2007ai) and SNL Section 7.1.2.2 (2007ak). DOE determined that the expected changes would be limited to small changes near the drift wall or, at a larger scale, within the range of variation that is already considered in the performance assessment. DOE reasoned that there would be little potential for cementitious materials in the repository to affect radionuclide transport by forming an alkaline cement leachate plume because (i) of the minor amount of cementitious material to be used in construction of the repository, none will be used in the waste emplacement drifts themselves and (ii) high pH conditions in an alkaline cement leachate plume would be rapidly neutralized in the unsaturated zone by reaction with ambient carbon dioxide. As a result, DOE concluded there would be little opportunity for the cement leachate to interact chemically with radionuclides or to affect radionuclide transport pathways by precipitation of calcite.

DOE also concluded that there would be little potential for evolved waste package fluids to cause more than minor changes in unsaturated zone fluid compositions. In DOE Enclosure 18 (2009ab) DOE cited the description of waste package chemistry (SAR Section 2.3.7.5.3.1) in stating that the main chemical factors in the effluent that affect radionuclide solubility will generally overlap the expected ranges of composition of the ambient unsaturated zone waters. Any change in effluent composition by reaction with the main chemical components of the engineered materials (iron, chromium, nickel) will be limited by the formation of low-solubility corrosion products inside the waste package. DOE reasoned that the waste package effluent may become concentrated by evaporation or consumption of water by degradation reactions, but upon exiting the waste package, the mixing of effluent with ambient waters in the invert and unsaturated zone would quickly dilute the effluent, resulting in no significant changes in bulk water chemistry in the unsaturated zone.

NRC Staff's Review

The NRC staff analyzed DOE's model assumptions, empirical data, and model support related to water–rock interactions at elevated temperatures and related to other geochemical interactions in the unsaturated zone influenced by reaction with cementitious engineering materials or waste package effluent. The geochemical modeling analyses DOE cited support DOE's explanation that changes in the unsaturated zone resulting from geochemical interactions at elevated temperatures, and those involving waste package effluent, are within the expected range of ambient conditions. Additionally, DOE's basis for excluding geochemical interactions with alkaline cement leachate is reasonable because (i) the repository design limits the use of cementitious materials near the waste emplacement drifts and (ii) geochemical interactions with carbon dioxide would neutralize the effects of an alkaline plume in the unsaturated zone. Therefore, DOE's consideration of potential geochemical processes, as summarized in the preceding paragraphs, provides a reasonable basis to exclude the feature, event, and process Geochemical Interactions and Evolution in the Unsaturated Zone on the basis of low consequence.

FEP 2.2.08.04.0A, Re-Dissolution of Precipitates Directs More Corrosive Fluids to Waste Packages

DOE excluded Re-Dissolution of Precipitates Directs More Corrosive Fluids to Waste Packages on the basis of low consequence (SNL, 2008ab). According to DOE's description of the feature, event, and process, the heat generated by radioactive decay inside the waste packages is expected to dry out the rock surrounding the emplacement drifts. Evaporation of the pore waters will leave behind precipitates that may plug pores. Re-dissolution of precipitates may produce a pulse of fluid reaching the waste packages when gravity-driven flow resumes, which is more corrosive than the original fluid in the rock (SNL, 2008ab).

DOE expects rewetting of the host rock around the drifts to occur as the temperature drops below the boiling point of water. Initially, DOE explained, precipitates could dissolve and form brines. During the initial stages of rewetting, re-dissolution of precipitated minerals may temporarily concentrate chloride and other soluble components relative to ambient solutions. As rewetting continues, DOE expects the brines to become diluted and pore waters to return to ambient compositions. DOE stated that the drip shield is expected to perform its diversion function during the time when the transient changes in pore water composition could occur, preventing potentially corrosive waters from contacting waste packages (SNL, 2008ab).

In addition to the undisturbed repository performance, DOE also evaluated this feature, event, and process in the event of early drip shield failure and seismic events. In the event of early drip shield failure, DOE assumed that a waste package under a compromised drip shield and at a seepage location would fail by localized corrosion; thus no additional failures would occur as a result of compositional changes due to re-dissolution of precipitates. In the event of a seismic event prior to rewetting and re-dissolution of precipitates, DOE described that the frequency and extent of drip shield failure would be generally insignificant. DOE also excluded FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, on the basis of low consequence, as described in SNL (2008ab) and DOE Enclosure 2 (2009ab). In that feature, event, and process, DOE analyzed in the screening justification scenarios allowing for water infiltrating failed drip shields and contacting the waste packages and concluded that those scenarios would not change the magnitude of the dose estimates, as identified in SNL (2008ab) and DOE Enclosure 2 (2009ab).

NRC Staff's Review

DOE's technical basis for exclusion is based on the drip shield performance. On the basis of the review documented in TER Section 2.2.1.3.2, the DOE conclusion that the drip shield would protect the waste package during a potential re-dissolution period is reasonable. The NRC staff also considered the DOE screening justifications for FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields. DOE provided a probabilistic analysis in DOE Enclosure 2 (2009ab) to estimate the additional number of waste packages failed and additional radionuclide releases if drip shields failed in the first 10,000 years due to seismic events. DOE concluded that additional consequences would be too small to change the dose estimates. As explained in the NRC staff's reviews of FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields, the DOE conclusion is reasonable and consistent with the review documented in TER Section 2.2.1.3.2. Therefore, exclusion of the feature, event, and process is reasonable on the basis of the drip shield function, which will prevent contact of potentially corrosive fluids with the waste packages during the thermal period when the potential for such conditions would exist. With respect to drip shield failure by seismic events in the first 10,000 years, DOE's conclusion that additional consequences would not affect dose estimates is reasonable. Past the thermal pulse period, the DOE abstraction predicts that there is a low probability for the repository environment (i.e., temperature, pH, and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package. The staff determined that DOE abstractions for the chemistry of water in the drifts and localized corrosion, evaluated in TER Sections 2.2.1.3.3.3.2 and 2.2.1.3.1.3.2.2 are reasonable. Thus, as related to this feature, event, and process and its technical basis for exclusion, there are no potential consequences from seismic events beyond the 10,000-year postdisposal period through the period of geologic stability. Therefore, the exclusion of the feature, event, and process Re-Dissolution of Precipitates Directs More Corrosive Fluids to Waste Packages from the performance assessment is reasonable.

FEP 2.2.09.01.0B, Microbial Activity in the Unsaturated Zone

DOE excluded Microbial Activity in the Unsaturated Zone on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE Enclosure 1 (2009ci). According to DOE's definition of the feature, event, and process, microbial activity may affect radionuclide sorption processes in the unsaturated zone by changing groundwater pH and redox conditions, by adding complexing agents to the water, or by changing the valence state of certain radionuclides by biotransformation. In addition, a microbe suspended in water may act as a biocolloid, facilitating the transport of radionuclides in the unsaturated zone by sorbing to the microbe itself. DOE also considered in DOE Enclosure 1 (2009ci) that increased microbial activity associated with condensation in the unsaturated zone during the early thermal period could affect the chemistry of water entering the drifts as seepage.

In the screening justification, DOE cited information provided in another excluded FEP 2.1.10.01.0A, Microbial Activity in the Engineered Barrier System (SNL, 2008ab), and in a supporting report, BSC Section 6 (2004aq), that evaluated the potential impact of microbial activity on drift chemistry. DOE stated in BSC Section 6.3 (2004aq) that although laboratory analyses have identified a diverse microbial population in Yucca Mountain tuff samples, the microbes are largely dormant under ambient conditions due primarily to constraints on the availability of nutrients and water in the unsaturated tuffs, as identified in BSC Section 6.4 (2004aq). DOE stated that any variation in radionuclide sorption coefficients that might be caused by changes in water chemistry due to microbial activity are within the existing range of the sorption coefficient distributions used in the performance assessment calculations. Similarly, DOE reasoned that the uncertain, small concentration of biocolloids in Yucca Mountain groundwaters is encompassed by the wide range of concentration values for naturally occurring colloids that is already sampled for radionuclide transport calculations. DOE also noted that the uncertainty distributions specified for the sorption coefficients in the performance assessment calculations implicitly included the effects of naturally occurring microbial activity because DOE developed the radionuclide K_d distributions from sorption experiments on the basis of the chemistry of Yucca Mountain water samples, as described in FEP 2.2.09.01.0B (SNL, 2008ab).

NRC Staff's Review

The NRC staff has reviewed the adequacy of DOE's sorption-related and colloid-associated unsaturated zone transport parameters in TER Section 2.2.1.3.7 and notes that the ranges of parameter estimates in the performance assessment are reasonable. On the basis of the NRC staff's evaluation of the site-specific data DOE used to support the unsaturated zone radionuclide transport abstraction, DOE's screening analysis for excluding microbial activity in the unsaturated zone adequately assessed the potential effects of microbial activity under ambient conditions.

For perturbed conditions in the unsaturated zone during the repository's early thermal period, DOE stated in DOE Enclosure 1 (2009ci) that microbial activity associated with water vapor condensation in the near-field rock beyond the dryout zone would not significantly affect seepage chemistry for three main reasons. First, a change from ambient to warmer temperatures in the near-field rocks would not increase the availability of already scarce nutrients in the rock, so the nutrient limitation on microbial activity would persist. Second, DOE estimated from modeling calculations that even for ambient conditions, the relative humidity of the densely welded Topopah Spring tuff at the repository horizon is already at the upper end of conditions that support optimal microbial activity in the matrix pore spaces. As a result, DOE reasoned that an increase in microbial activity in the condensation zone would be limited primarily to water that condensed in fractures. Third, DOE stated that for the near-field environment as a whole, any increased microbial activity in the condensation zone would be offset by reduced microbial activity in the dryout zone during the same period.

The NRC staff reviewed DOE's screening justification for microbial activity during the thermal period in the context of DOE's thermohydrologic models for Yucca Mountain, as described in SNL Section 6.1 (2008aj); site characterization data for the densely welded Topopah Spring tuff, described in BSC Section 7.2 (2004bi); Yucca Mountain field tests and interpretations of fracture flow and transport processes in BSC Section 7.6.3.1 (2004bi) and BSC Section 6.2.4 (2006aa); and observations of increased microbial activity in fractures during an unsaturated zone infiltration and transport experiment at Yucca Mountain, explained in BSC Section 6.1.2 (2006aa).

The NRC staff has reviewed DOE's thermohydrologic models in TER Section 2.2.1.3.6 and notes they provide a basis for the performance assessment abstraction. DOE's information on increased microbial activity in the unsaturated zone during the thermal period in DOE Enclosure 1 (2009ci) identified that scarcity of nutrients in the rocks is one of the main factors limiting microbial activity. DOE has provided site characterization data to support the statement that low concentrations of specific nutrients, such as phosphate and organic carbon, are expected to persist in the subsurface environment throughout the postclosure period for ambient as well as thermal conditions. DOE appropriately concluded that microbial activity would continue to be restricted by nutrient availability during the thermal period because there are no significant processes that would enhance the concentration of nutrients in the rocks at elevated temperatures compared to the ambient concentrations.

DOE stated that despite the limited availability of nutrients, the increased availability of water in fractures may contribute to an increase in microbial activity in the condensation zone. The technical basis for DOE's statement is supported by observations from a Yucca Mountain large-scale unsaturated zone field experiment, the Large Plot Test, which reported in BSC Section 6.1.2 (2006aa) the growth of biofilms (slime layers excreted by certain microbes at solid–water interfaces) in fractures on a wetted floor in the subsurface at Yucca Mountain. DOE provided a reasonable technical basis for the statement that increased microbial activity in the condensation zone would be limited to water in the fractures because water evaporated from the dryout zone would migrate preferentially through fractures and condense in cooler locations outside the dryout zone. DOE's technical basis stated that the water in fractures represented only a small fraction of the total water content of the rock, but DOE's technical basis did not address the potential effect of increased microbial activity in the condensation zone fractures in the context of (i) the importance of water in fracture-dominated flow pathways as potential seepage into drifts (SAR Section 2.3.3.1) or (ii) the potentially large fraction of water transferred by evaporation from the dryout zone to the fractures of the condensation zone during the thermal period, as detailed in SNL Section 6.1.2 (2008aj). However, DOE's conceptual thermohydrologic model, which was described in SNL Section 6.1.3 (2008aj), shows that during the thermal period, the water condensed in the fractures does not remain there but continually drains downwards and away from the dryout zone along a network of connected fractures between the emplacement drifts. The NRC staff notes that DOE's analysis supports the DOE conclusion that little of the microbially affected water would remain in fractures above the drifts by the time ambient flow returns to the repository near-field rocks after the thermal period.

DOE also stated an increase in microbial activity in the condensation zone would be offset by restricted microbial activity in the dryout zone, so there would be no net increase in microbial activity during the thermal period. This explanation is supported by information from microbial studies that suggest elevated temperatures in the dryout zone would restrict microbial activity at that location during the thermal period (DOE, 2009ci). The NRC staff notes that the DOE-provided information supports the DOE conclusion that the increased availability of water for microbial activity in the condensation zone would be balanced by the loss of the same amount of water from the dryout zone.

In summary, DOE provided a technical basis that showed that increased microbial activity in the unsaturated zone would not significantly affect the chemistry of water seeping into repository drifts, because (i) the limited availability of nutrients would restrict microbial activity in both ambient and thermal conditions and (ii) only a fraction of the water potentially affected by increased microbial activity in the condensation zone during the thermal period would be available as seepage to the repository drifts when ambient flow paths were restored. DOE provided a reasonable technical basis to exclude the feature, event, and process Microbial Activity in the Unsaturated Zone on the basis of low consequence.

FEP 2.2.10.09.0A, Thermo-Chemical Alteration of the Topopah Spring Basal Vitrophyre

DOE excluded Thermo-Chemical Alteration of the Topopah Spring Basal Vitrophyre on the basis of low consequence (SNL, 2008ab). According to DOE's information, the Topopah Spring basal vitrophyre is a glassy, densely welded tuff that forms the lowermost part of the Topopah

Spring tuff hydrogeologic unit. DOE used the feature, event, and process to examine the possibility that temperatures elevated by repository heating might cause the volcanic glass in the vitrophyre to alter to zeolites and clay minerals, potentially changing permeability and flow paths in the basal vitrophyre, and increasing the sorptive properties of the unit (SNL, 2008ab).

In the screening justification, DOE stated that although heat from the repository will locally increase temperatures in the unsaturated zone for hundreds to several thousand years, the potential for any thermo-chemical alteration of the vitrophyre would be of limited spatial extent and of short duration compared to the previous alteration history of the Topopah Spring tuff. DOE cited fluid inclusion and isotope studies of fracture minerals in the Topopah Spring tuff units to support the screening argument. The studies identified that regionally elevated temperatures {above 80 °C [176 °F]} occurred in the tuffs at least 10 million years ago, followed by gradual cooling over several million years to near-ambient conditions. Despite its long exposure to the elevated temperatures, the basal vitrophyre remained largely unaltered. DOE reasoned that, by comparison, the relatively brief and spatially limited postclosure thermal pulse from the repository will only minimally alter the vitrophyre to secondary minerals, so that any effects on the sorptive or hydrologic properties of the unit will not result in a significant adverse change in the magnitude or timing of either radiological dose to the reasonably maximally exposed individual or radionuclide release to the accessible environment.

NRC Staff's Review

The NRC staff examined DOE's cited modeling studies, empirical data, and glass alteration rates (SNL, 2008ab) that supported DOE's representation of the past thermal history of the Topopah Spring tuff units and that supported DOE's thermal model predictions for repository heating. On the basis of the information provided, DOE supported the screening argument that repository heating will not cause significant thermo-chemical alteration of the Topopah Spring basal vitrophyre. Because thermo-chemical alteration of the Topopah Spring basal vitrophyre will only minimally alter the vitrophyre and the effects on flow and sorption will be small, any changes in the sorptive or hydrologic properties of the unit would be of low consequence for repository performance. DOE's technical basis to exclude the feature, event, and process from the performance assessment on the basis of low consequence is reasonable.

FEPs 2.1.14.15.0A Through 2.1.14.26.0A and 2.2.14.09.0A Through 2.2.14.12.0A, Criticality Features, Events, and Processes

The criticality features, events, and processes encompass FEPs 2.1.14.15.0A through 2.1.14.26.0A and 2.2.14.09.0A through 2.2.14.12.0A (SAR Table 2.2-5) and are classified as events. DOE excluded all of the criticality features, events, and processes from the performance assessment on the basis of low probability (SAR Table 2.2-5; SNL, 2008ab). As described in SAR Section 2.2.1.1.2 the potential for criticality events is determined by a number of precursor conditions that must occur for the inventory to achieve a potentially critical configuration. An initiating event must occur that causes a breach of the waste package before any other sequence of events on that waste package could lead to criticality. Additional precursor conditions include (i) presence of a moderator (i.e., water), (ii) separation of fissionable material from the neutron absorber material or an absorber material selection error during the canister fabrication process, and (iii) the accumulation (external) or presence of a critical mass of fissionable material in a critical geometric configuration. As described in SNL (2008ab), the probability of developing a configuration with criticality potential is insignificant unless the initiating event and all three of the precursor conditions are realized.

DOE's technical basis for exclusion of the criticality features, events, and processes consisted of a probability analysis based on location, initiating events, and state of degradation (e.g., the waste package internal structures and the waste form may degrade). DOE divided the criticality features, events, and processes into three locations for four initiating event scenarios, as described in SAR Section 2.2.1.4.1: internal to the waste package in an intact or degraded condition, near field, and far field. DOE used four initiating event scenarios: nominal, seismic, rockfall, and igneous. DOE described that these scenarios are different from the scenario classes described in SAR Section 2.2.1.3 because the scenario classes described in SAR Section 2.2.1.3 were formulated for analyses of included events in the performance assessments. DOE's analysis of the probability of criticality classified the waste forms by type—commercial spent nuclear fuel (CSNF), high-level radioactive waste glass, and DOE spent nuclear fuel (SNF)—which were further subdivided, in some cases. As described in SAR Section 2.2.1.4.1, analyses of the in-package probability of criticality for naval spent nuclear fuel are described in the Naval Nuclear Propulsion Program Technical Support Document (*classified*) Section 2.2.1.4.1 and subsequent discussions of near- and far-field criticality in SAR Section 2.2.1.4.1 are applicable to naval spent nuclear fuel as well as all other waste form types.

In TER Section 2.2.1.2.1.3.3, the NRC staff reviews formation of scenario classes. The formation of scenario classes refers to the aggregation of features, events, and processes into event classes or scenario classes for the purpose of further screening or analyses. As described in SAR Section 2.2.1.3, DOE aggregated criticality events into the criticality event class. SAR Section 2.2.1.4.1 described DOE's nuclear criticality considerations for the repository during the postclosure period and reviewed the technical basis by which the nuclear criticality event class is screened from the postclosure performance assessment on the basis of low probability.

DOE's technical basis for screening out the individual criticality features, events, and processes was provided in SNL (2008ab). DOE supplemented its technical basis for exclusion of individual features, events, and processes and, in some cases, the technical basis for exclusion of the criticality event class in DOE Enclosures 1–18 (2009aj), DOE Enclosures 1–11 (2009al), DOE Enclosures 1–5 (2009gp), DOE Enclosures 1–2 (2009gq), DOE Enclosures 1–10 (2009bv), DOE Enclosures 1–11 (2009bw), DOE Enclosures 1–2 (2009bx), DOE Enclosures 1–4 (2009co), DOE Enclosures 1–2 (2010ah), DOE Enclosures 1–2 (2010ad), and DOE Enclosures 3–5 (2009cb).

In this section the NRC staff reviews DOE's technical basis for excluding the individual criticality features, events, and processes and the technical basis for excluding the criticality event class. The NRC staff reviewed the information provided in the application, including the cited references, and noted that the in-package intact configuration and in-package degraded configurations are the most risk significant configurations. As a result, these configurations were evaluated in greater detail.

Criticality features, events, and processes that consider igneous events (FEPs 2.1.14.24.0A through FEP 2.1.14.26.0A and FEP 2.2.14.12.0A) were not evaluated in detail, because the igneous events have a low probability of occurrence (TER Sections 2.2.1.2.2 and 2.2.1.3.10). When the probability of the igneous events are combined with other low probability physical requirements for a critical configuration to occur, the total combined probability is negligible.

For the near- and far-field criticality features, events, and processes (FEP 2.1.14.17.0A, FEP 2.1.14.20.0A, FEP 2.1.14.23.0A, FEP 2.1.14.26.0A, and FEPs 2.2.14.09.0A through

FEP 2.2.14.12.0A), DOE adopted a number of assumptions to conservatively overestimate the probability of criticality. The two most significant assumptions were (i) fissile material would accumulate in the optimum geometry for criticality and (ii) there would be no nearby neutron absorbers or fission products. Despite these assumptions, near- and far-field criticality is a negligible contributor to the overall probability of criticality [see SAR Table 2.2-8; the cutoff for including probabilities in this table is two orders of magnitude lower than 10^{-8} per year]. Therefore, the NRC staff did not perform a detailed review of the near- and far-field criticality features, events, and processes.

DOE's design basis configuration model incorporates some rearrangement and degradation of fuel and neutron absorbers to overestimate the probability of a criticality event. DOE concluded that under such configurations, a criticality event could occur only if the waste package was misloaded or if a manufacturing error resulted in missing neutron absorbers, as identified in FEP 2.1.14.15.0A (SNL, 2008ab). On the basis of the NRC staff's review of the information provided in the application, including the cited references, on the in-package intact configuration and in-package degraded configurations, the NRC staff focused its review on the following aspects of the DOE criticality analysis that are important in estimating the probability of criticality and the conditions that are inputs into and that might limit the applicability of the probability analysis:

- Moderator intrusion
- Misloaded fuel
- Neutron absorbers
- Burnup credit
- Criticality code validation

The following subsections summarize DOE's approach and the NRC staff's evaluations for each of these aspects of the DOE technical basis and conclusions.

Moderator Intrusion

The presence of a moderator is necessary for the spent fuel to go critical. Because water is a neutron moderator, its presence results in increased reactivity. Unbreached waste packages do not allow ingress of water (neutron moderator) and thus do not pose a criticality concern. DOE conservatively assumed that enough water is available to act as a neutron moderator in the criticality calculations whenever any breach of the package is calculated—even for cracks on the waste package that are too small to permit liquid infiltration (SNL, 2008ab). No credit is taken for only partial filling of the waste package, nor for the drip shield's diversion of liquid from the package (DOE, 2009bx); thus the waste packages, once breached, are assumed to be flooded. Because DOE assumed that failed waste packages would be flooded (SNL, 2008al), the probability of criticality is overestimated. In the nominal modeling case, DOE estimates that the waste packages would not be breached in 10,000 years (e.g., SAR Figure 2.1-10). However, other modeling cases (early failure, seismic ground motion, seismic fault displacement, and igneous intrusion) account for failure of the waste package within 10,000 years (e.g., SAR Figure 2.4-18). Because of the low probability for igneous intrusion and fault displacement, DOE concluded that criticality events could only occur within 10,000 years for the early failure and seismic scenarios. In its criticality analysis, DOE considered various reactivity control mechanisms within a waste package (e.g., neutron absorbers) to ensure that all probable configurations remain subcritical.

NRC Staff's Review of Moderator Intrusion

The NRC staff evaluated (i) whether the probability calculation was performed correctly, (ii) the appropriateness of the inputs used in the calculation, (iii) consideration of uncertainties, and (iv) DOE's determination that the probability of the criticality features, events, and processes is less than 1 chance in 10,000 of occurrence within 10,000 years of disposal. The NRC staff reviewed documents with analyses supporting screening arguments for excluding criticality events from the performance assessment analysis (SNL, 2008aa,ab,ad,ae,al) and supplemental analyses in DOE Enclosures 1–18 (2009aj), DOE Enclosures 1–11 (2009al), DOE Enclosures 1–5 (2009gp), DOE Enclosures 1–2 (2009gq), DOE Enclosures 1–10 (2009bv), DOE Enclosures 1–11 (2009bw), DOE Enclosures 1–2 (2009bx), DOE Enclosures 1–4 (2009co), DOE Enclosures 1–2 (2010ah), DOE Enclosures 1–2 (2010ad), and DOE Enclosures 3–5 (2009cb). DOE assumed that given a postulated breach of the spent fuel package, no matter how small a breach, the waste package would fill with enough water to support criticality (i.e., the availability of water was not assumed to be a limiting factor for criticality). This assumption is conservative because it results in an overprediction of the potential for criticality. It allows for enhanced neutron moderation compared to the much more probable situation of limited water ingress into the waste package, especially for those waste packages breached by very small cracks.

The NRC staff notes that DOE adequately identified and quantified conditions that could lead to in-package criticality. The NRC staff reviewed the conditions DOE identified as necessary for in-package criticality, as well as the methodology used to identify these conditions. The total probability of criticality is dependent on the probability to attain those necessary conditions (e.g., waste package failure, moderator intrusion, configuration changes, package/absorber degradation, and fuel characterization). DOE stated that a criticality event was possible only after a waste package breach and under the following conditions: (i) accidentally loading a fuel assembly with higher reactivity than permissible into the waste package (a mistake referred to as a misload) or (ii) manufacturing errors resulting in missing neutron absorber material. The misloading criteria and manufacturing performance criteria used in the analysis are reasonable, because DOE applied human reliability data developed by the industry and the international community. The data appropriately considered dependencies and the human factors in manufacturing and loading procedures. Waste packages are safety-related equipment (preclosure) and important to waste isolation equipment (postclosure). Therefore, their manufacturing process requires stringent quality assurance programs.

The performance numbers are consistent with industry practices and performance when applying those quality assurance processes. DOE's quality assurance program is intended to complement the performance criteria credited for the waste package, the manufacturing and loading of neutron absorbers, and the loading of fuel assemblies, so that these factors can be relied on as part of the technical basis for exclusion of the criticality features, events, and processes and the criticality event class.

DOE identified in SAR Sections 1.5.1.4.1.2.6.3 and 2.2.1.4.1 that analyses of the in-package probability of criticality for naval spent nuclear fuel (SNF) are described in the Naval Nuclear Propulsion Program Technical Support Document (*classified*) Section 2.2.1.4.1. However, DOE summarized (*nonclassified*) aspects of the postclosure criticality analysis for naval spent nuclear fuel in SAR Sections 1.5.1.4.1.2.2.2, 1.5.1.4.1.2.5.2, 1.5.1.4.1.2.6.1, 1.5.1.4.1.2.6.3, and 2.2.1.4.1. As described in SAR Section 1.5.1.4.1.2.2.2, criticality control of naval spent nuclear fuel (i.e., assurance of a low probability that criticality involving naval spent nuclear fuel could occur) is provided by controlling one or more of the following characteristics of the loaded naval spent nuclear fuel canister: the amount of fissile material; the materials used for naval spent nuclear fuel canisters, baskets, spacers, naval corrosion-resistant cans, control rods, and installed neutron poison assemblies and their retention hardware; and geometric separation of naval spent nuclear fuel assemblies. As identified in SNL Table B-1 (2008al), naval waste packages are subject to the same breach scenarios as other packages. The NRC staff reviewed the DOE information and determined that the assumptions used in the models were conservative and appropriately applied. The NRC staff notes that DOE considered the parameters important to criticality for naval fuel and conservatively or realistically represented them in the screening calculation, which resulted in a 10,000-year probability of criticality (7.1×10^{-6}; SAR Table 2.2-8)—well below the screening criteria. Therefore, DOE has reasonably screened out in-package criticality for naval spent fuel on the basis of low probability.

Misloaded Fuel

DOE presented loading curves for commercial spent nuclear fuel in SNL Figures 6-32 and 6-33 (2008aa) and SAR Figures 2.2-7 and 2.2-8 and defined "acceptable" and "underburnt" assemblies that could be loaded into waste packages based on minimum burnup, as a function of initial fuel assembly enrichment. DOE considered assemblies above the loading curve acceptable, while those below the loading curve were considered underburnt. A canister filled with underburnt assemblies could exceed the critical limits listed in SAR Table 2.2-11(or become critical), if flooded, without additional criticality control mechanisms. Typographical errors in rows one and three in SAR Table 2.2-11 were corrected in DOE Enclosure 10 (2009bv), and DOE stated it would correct SAR Table 2.2-11. These underburnt assemblies comprise the potential misload inventory. Although DOE did not specify what additional reactivity control mechanisms or analysis will be used to meet the critical limit, DOE stated that the underburnt assemblies would have to be loaded into canisters with additional reactivity control mechanisms (e.g., disposal control rod assemblies) and individually analyzed to ensure subcriticality (SAR Section 2.2.1.4.1.1.3).

The waste package configuration used to compute the loading curves was selected to bound degraded configurations that were not explicitly evaluated in the screening argument [e.g., the conversion of UO_2 into schoepite, as described in SNL Sections 6.1.3 and 6.2.5 (2008aa)]. DOE considered it safe (i.e., criticality events are not expected) to load canisters with spent fuel whose burnup exceeds the minimum burnup defined by the loading curve for the initial enrichment of the fuel. DOE implemented these bounding analyses to provide confidence that waste package configurations not explicitly analyzed are less reactive than those that were analyzed and used to generate the loading curve.

DOE defined a misload as the process of loading, by mistake, a fuel assembly into a canister without enough criticality prevention controls for that fuel assembly (specifically, commercial spent nuclear fuel, when an assembly from the misload inventory is loaded into a canister). DOE assumed that a misload may cause a criticality event if the misloaded assembly is significantly more reactive than accounted for in the waste package design. DOE assumed misloads result from operator error and used representative human reliability data and prototype loading procedures to estimate the probability of misloads, because actual data and procedures are not available (DOE, 2003aa).

DOE assumed that misloads will not occur for DOE spent nuclear fuel, because the physical differences in fuel types allow operators to easily distinguish spent fuel types and in some cases physically prevent misloads [SNL, 2008ab; FEP 2.1.14.15.0A, In-package Criticality (Intact

Configuration)]. A commercial spent nuclear fuel misload would occur if an underburned assembly is loaded; however, a criticality event would only be possible if the other assemblies were not burned enough to compensate for the underburned assembly.

NRC Staff's Review of Misloaded Fuel

The NRC staff reviewed DOE's determination of the misload probability, which is one of the inputs into the overall probability of criticality (4.4×10^{-5}; SAR Table 2.2-8). DOE modeled all the misloads as random human error in the selection of fuel assemblies to be loaded, because human error is the dominant cause of misloads. This calculation resulted in a combined misload probability of 1.18×10^{-5} per canister. This is the probability that the operator will misload a single assembly into a canister.

DOE also calculated an additional probability related to misloads: the probability that a criticality event would occur given one assembly has been misloaded into a canister. As described in DOE Enclosure 13 (2009aj) the probability of criticality due to an assembly misload in a breached Pressurized Water Reactor (PWR) waste package is the product of the probability of a misload (1.18×10^{-5}) and the conditional probability of criticality given a misload (0.014). To calculate the conditional probability of criticality given a misload, DOE assumed that misloaded assemblies would be positioned in the center (i.e., along the axis) of the waste package. This assumption is reasonable because this position is the most reactive due to neutron interaction with the surrounding assemblies, maximizing the likelihood of criticality. In other words, this assumption is conservative with regard to estimates of the probability of criticality. However, DOE's analysis did not account for the probability of criticality if there are two or more misloads. Multiple misloads are likely to result from a common mode failure and cannot be accurately modeled as random misloads. In DOE Enclosure 13 (2009aj), DOE supplemented its previous analysis by performing a sensitivity study that assumed a criticality event resulted whenever a misload occurred (which could be a single assembly misloaded or the first of several assemblies being misloaded). Thus this analysis changed the value of the conditional probability of criticality from a misload from 0.014 to 1. DOE identified that this change in the conditional probability only causes the overall probability of criticality to increase to a value of 5.76×10^{-5} in 10,000 years, which is less than the exclusion criterion. Because in this sensitivity study every misload is assumed to cause a criticality event, the probability of criticality given multiple misloads is bounded. On the basis of this sensitivity study, DOE's conclusion that multiple misload assemblies alone would not cause criticality to exceed a probability of 10^{-4} in 10,000 years is reasonable.

Neutron Absorbers

To model the effects of missing neutron absorbers, DOE reduced the amount of absorber in the models. It assumed that loss of absorber results from manufacturing error or corrosion of neutron absorber materials. For the corrosion of the absorber plates in the transportation, aging, and disposal canister, DOE assumed that after 10,000 years, 6 mm [0.24 in] out of the initial 11 mm [0.43 in] of borated stainless steel would remain in place. In response to the NRC staff's RAIs on use of average corrosion values in DOE Enclosures 1 and 2 (2009bv) and in SAR Section 1.5.1.1.1.2.2.2, DOE indicated that the 5 mm [0.2 in] material thinning was based on a borated stainless steel corrosion rate of 250 nm/yr [0.01 mil/yr]. This corrosion rate is about nine times the average corrosion rate on 304B4 borated stainless steel Lister, et al. (2007aa) measured. Lister, et al. (2007aa) measured the corrosion rate at 60 °C [140 °F] in an aerated simulated in-package solution and determined an average value of 27 nm/yr [0.001 mil/yr] with a standard deviation of 10.1 nm/yr [4×10^{-4} mil/yr]. Although some boron would remain in the steel as separate chromium boride particles left behind as insoluble products during corrosion, this remaining boron was not credited in DOE's criticality models. In its criticality models with the SCALE and MCNP computer codes, DOE modeled 75 percent of the boron that exists in the stainless steel as per the guidance in NUREG–1567 (NRC, 2000ab).

DOE assumed that if a manufacturing error resulted in neutron absorbers not being installed or too little absorber material being installed, a criticality event could occur. DOE used a surrogate analysis to model the probability of this error occurring as the probability of a material selection error multiplied by the probability that an independent inspection does not detect it to derive a mean value of 1.25×10^{-7} per canister. DOE considered representative reliability data and prototype manufacturing procedures to develop an overestimate of the probability of misloading the neutron absorber plates.

The naval waste packages use hafnium (SAR Section 1.5.1.4.1.2.2.2)—a strong thermal neutron absorber that is extremely corrosion resistant (Rishel, et al., 2000aa)—as a neutron absorber. For the absorbers considered in the DOE spent nuclear fuel canisters, DOE evaluated the solubility, retention, and distribution of the neutron absorbers in DOE Enclosure 5 (2009cb). DOE has not yet completed the design of the neutron-absorbing shot that will be added to some waste forms (waste forms DOE referred to as DOE1, DOE5, and DOE8). DOE stated in DOE Enclosure 5 (2009cb) that due to its high corrosion resistance, $GdPO_4$ is the most likely form of neutron absorber.

NRC Staff's Review of Neutron Absorbers

The NRC staff's review of DOE's SAR and supporting documents related to the neutron absorber design and performance notes that potential degradation of neutron absorbers was reasonably addressed in the criticality analyses. The NRC staff reviewed the corrosion rates of borated and nonborated stainless steel reported in the literature, such as Beavers and Durr (1991aa); BSC (2004ae); Beavers, et al. (1992aa); McCright, et al. (1987aa); Glass, et al. (1984aa); Fix, et al. (2004aa); and Lister, et al. (2007aa,ab). The NRC staff noted that in fresh water, J-13, simulated J-13, J-13 with crushed tuff water, and simulated concentrated waters in a temperature range of 28−100 °C [82.4−212 °F], 304 and 316 stainless steels have similar general corrosion properties in these solutions, but if borated, the corrosion rate of stainless steel increases. The information in Fix, et al. (2004aa), BSC Tables 6-4 and 6-7 (2004ae), and SNL (2007am) indicated that the corrosion rates of borated stainless steels were higher than for unborated 304 and 316 stainless steels. BSC (2004ae) also indicated that the corrosion rate of borated 304 stainless steel with 1.5 percent boron was about 14 times higher than that with 0.3 percent boron in boiling freshwater. The corrosion rates of borated stainless steel ranged from tens of nanometers per year (Lister, et al., 2007aa) to tens of micrometers per year in BSC Table 6-7 (2004ae), depending on simulated environmental conditions.

The NRC staff reviewed the data and the applicable environmental conditions that DOE considered credible to reside within the waste package and surroundings. High corrosion rates in the range of micrometers per year were obtained from immersion tests. These high corrosion rates may lead to thinning of the borated stainless steel to below the 6-mm [0.24-in] thickness DOE considered. However, the NRC staff noted that those high corrosion rates are unlikely and need no further consideration in DOE's criticality analysis. These high corrosion rates are unlikely because they (i) require full water immersion conditions and (ii) require the presence of ionic species that only seepage water can supply. Given that the extent of damage of the waste packages that could fail in 10,000 years (waste packages could fail due to early failure, due to localized corrosion if under early failed drip shields, or due to stress corrosion cracking) is limited and that drip shield failure is required to allow seepage water ingress into the waste package, the NRC staff noted that sufficient ingress of the ionic species present in seepage water to maintain a water chemistry needed to support high corrosion rates is unlikely.

Because liquid water seepage into a waste package is unlikely, water ingress into the waste packages in 10,000 years will most likely be in the form of water vapor. Accordingly, the NRC staff reviewed literature data on borated stainless steel corrosion under humid air conditions. Beavers and Durr (1991aa) studied corrosion of 304 stainless steel under vapor and aqueous conditions at 90 °C [194 °F] and concluded that corrosion rates were under their detection limit of 200 nm/yr [7.9×10^{-3} mil/yr]. The corrosion rate value DOE used to support the use of a 6-mm [0.24-in]-thick plate of borated stainless steel in the criticality computations, 27 nm/yr [0.001 mil/yr], is below the upper bound of 200 nm/yr [7.9×10^{-3} mil/yr] estimated by Beavers and Durr (1991aa). No precise estimates of corrosion of borated stainless steel under vapor are available. The NRC staff noted, however, that even if the corrosion rate under humid air exceeded 27 nm/yr [0.001 mil/yr], stainless steel is not expected to continuously corrode from the time of closure to 10,000 years after closure {as assumed in DOE's derivation of the 6-mm [0.24-in]-thick value}, because the waste packages could fail at different times in the 10,000-year period and the spent fuel is a heat source that could mobilize water away. Also, boron present in corrosion products could still be an effective neutron absorber. Accordingly, use of a 6-mm [0.24-in]-thick plate of borated stainless steel in the criticality analysis is adequate.

DOE's incorporation of a degraded neutron absorber in its criticality analyses is reasonable. The NRC staff notes that the neutron absorbers in the transportation, aging, and disposal canister, with the assumption that neutron absorber plates are at least 6 mm [0.24 in] thick for 10,000 years, provide adequate reactivity control. Given the stricter quality assurance requirements DOE proposed for components that are classified as important to safety and important to waste isolation, DOE's quality assurance program for design and construction of the neutron absorber plates will provide reasonable basis that the probability of a manufacturing error was adequately evaluated.

With respect to the naval waste packages, DOE's analysis of the system reactivity considering the presence of hafnium as a neutron absorber material is reasonable because of its well-known corrosion resistance properties and the Navy's long experience with its use in reactors to control criticality.

The NRC staff reviewed DOE's proposed use of a gadolinium absorber shot, which prevents criticality for some of the most reactive DOE spent nuclear fuel types. The NRC staff notes that filling the DOE spent nuclear fuel canisters with enough absorber shot to keep the most reactive intact or degraded configuration subcritical, with an adequate margin, is a reasonable use of neutron absorbers. This is because DOE models both degraded and undegraded absorber shots, which provided the NRC staff's confidence that the absorber shot will perform its intended function.

Burnup Credit

DOE uses the burnup of the commercial spent nuclear fuel to control criticality in much the same way that neutron absorber plates are used. However, unlike absorber plates, the amounts of absorbers (and fissile material) in each assembly vary and must be computed analytically. DOE accounted for the change in reactivity caused by changes in fuel composition that resulted from irradiation in a reactor and radioactive decay. This is mostly due to the buildup of fission products that are neutron absorbers and to the depletion of fissile material, although some fissile material is also created. To compute the change in reactivity, DOE modeled commercial spent nuclear fuel as being composed of 29 principal isotopes (SAR Table 2.2-9) considered to be the most relevant and concluded in DOE Section 6 (2004ab) that increasing the burnup of commercial spent nuclear fuel decreases its reactivity. As described in SAR Section 2.2.1.4.1.1.2.4.1, DOE validated the isotopic model by comparing the results of the model to the results of radiochemical assays that measured the amount of some or all of the principal isotopes in small samples of commercial spent nuclear fuel (DOE, 2004ab).

DOE used reactor records to determine the burnup of the fuel (SAR Section 2.2.1.4.1.1.4.1) and a computer model to generate the isotopic composition for a given burnup and enrichment. DOE accounted for the uncertainties in the reactor records by using a burnup that is 5 percent less than reported. This adjustment bounded the highest reactor record uncertainty identified in AREVA (2004), which was 4.2 percent. The reactor record uncertainty was determined from the difference between the calculated and measured values of burnup. Measured burnup was determined with calibrated in-core instrumentation. Calculated burnup was determined using analytic methods of the reactor core power distribution. Uncertainties in the isotopic composition were accounted for by using modeling parameters that would conservatively overestimate the reactivity of the fuel and thus overestimate the probability of criticality (SAR Section 2.2.1.4.1.1.2.4.1). In DOE Enclosure 2 (2009bx), DOE committed to using a maximum burnup of 50 GWd/MTU with respect to the burnup credit loading curves in criticality analyses for commercial pressurized water reactor fuel unless additional validation data are provided to extend the burnup beyond 50 GWd/MTU. DOE used a decay time for the isotopic composition of 5 years to bound the increase in reactivity caused by Am-241 and Pu-240 decay.

For boiling water reactor (BWR) spent nuclear fuel, DOE performed two different criticality analyses. One analysis applies to BWR spent nuclear fuel with initial enrichments up to 4.5wt% U-235; this analysis does not incorporate burnup credit in the analysis. For the second analysis, some BWR burnup credit is necessary to accommodate fuel assemblies with higher initial enrichments (i.e., > 4.5wt% U-235) than the projected waste stream inventory in SNL Figure 6-33 (2008aa). As discussed in DOE Enclosure 2 (2010ah), only a small amount of burnup credit is required for disposal; in other words, only a fraction of the burnup needs to be credited. While this analysis was performed with a “simple” BWR assembly design as compared to the “modern” and more advanced fuel designs, the more advanced designs ultimately result in a net decrease in the total fissile mass available in the fuel assembly at discharge. DOE stated that assembly designs that have not been explicitly analyzed (like the advanced BWR designs) are required to be evaluated to ensure appropriateness from a postclosure criticality perspective in accordance with SAR Section 5.10.2 and SAR Table 5.10-3. As stated by DOE in DOE Enclosure 2 (2010ah), SAR Table 5.10-3 described a number of administrative controls to be included in license specifications, and these administrative controls would be compiled in the Technical Requirements Manual.

NRC Staff’s Review of Burnup Credit

The NRC staff reviewed DOE’s use and justification of BWR fuel burnup credit. For BWR fuel assemblies with enrichment below 4.5 wt% U-235, DOE’s analytic methodology that does not incorporate burnup credit is reasonable. However, a very small percentage of BWR fuel assemblies may contain enrichment greater than 4.5wt% U-235. For these assemblies, DOE requested burnup credit. While DOE did not have the requisite data to support the isotopic composition and associated reactivity of these fuel assemblies (DOE, 2004ab), the NRC staff does not consider DOE’s inability to accurately predict the isotopic composition and reactivity of modern BWR spent nuclear fuel to be risk significant. This is because DOE uses conservative input parameters to the irradiation and depletion calculations when generating the BWR isotopic database spent fuel isotopic compositions with a small amount of burnup being credited. As a result, DOE only takes credit for a portion of the burnup credit from the principal isotopes that are expected to be available. Additionally, staff does not consider granting BWR burnup credit to be risk significant, because the administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses (similar to the analysis performed for the simple BWR design) be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the SAR. Therefore, the NRC staff considers it reasonable for DOE to take burnup credit for BWR spent nuclear fuel. The NRC staff reviewed DOE's use and justification of pressurized water reactor burnup credit. DOE used conservative input parameters (such as fuel types and operating histories) in the irradiation and depletion calculations when generating the isotopic composition of the pressurized water reactor fuel. Taking full credit for the neutron absorptive properties of Mo-95, Tc-99, Ru-101, Rh-103, and Ag-109 was technically unjustified due to insufficient and inadequate radiochemical assay data, which were primarily based on the 11 TMI-1 radiochemical assay samples. However, in DOE Enclosure 1 (2010ah), DOE showed that the isotopic bias and uncertainty incorporated into the critical limit should make up for the errors and uncertainties in the predictions of these five isotopes. Furthermore, the presence of some neutron-absorbing fission products that DOE did not credit in the analysis, the radiochemical assay data from a variety of reactors, and conservative input parameters provide staff's confidence that the overall uncertainties in the isotopic predictions of the 29 principal isotopes for pressurized water reactor burnup credit are not sufficient to cause a significant increase in the probability of criticality.

Criticality Code Validation

In SAR Section 2.2.1.4.1.1.2.4, DOE described that its validation process used for the criticality model is summarized in four steps: (i) selection of benchmark experiments, (ii) establishment of the range of applicability of the benchmark experiments, (iii) extension of the range of applicability (as necessary), and (iv) development of critical limits. DOE described that its validation methodology was performed in accordance with ANSI/ANS 8.1-1983, Section 4.3 and Appendix C (American Nuclear Society, 1983aa). DOE also described that Regulatory Guide 3.71 (NRC, 1998aa) endorses the use of ANSI/ANS-8 nuclear criticality safety standard documents and states that the procedures and recommendations in the ANSI/ANS-8 standards should be followed to prevent and mitigate nuclear criticality event sequences.

An important aspect in assessing criticality code validation is the applicability of the selected benchmark experiments, which must be similar in form and composition to the systems to be evaluated. DOE described the benchmarks used for the criticality code validation, and the determination of the lower bound tolerance limits for commercial spent nuclear fuel and DOE spent nuclear fuel, in DOE (2004aa) and Radulescu, et al. (2007aa). DOE included in the criticality model document (DOE, 2004aa) analyses for the various types of commercial and DOE spent nuclear fuel. The DOE updated its criticality validation methodology and benchmarks for commercial spent nuclear fuel in Radulescu, et al. (2007aa) as explained in DOE Enclosure 7 (2009al). DOE has not yet updated the benchmark selection and validation for DOE spent nuclear fuel with the new methodology used in Radulescu, et al. (2007aa). In the note to SAR Table 2.2-11, DOE stated DOE spent nuclear fuel interim critical limits have not been rigorously established for all fuel groups, but will be confirmed prior to waste acceptance as identified in Table 5.10-3. However, in DOE Enclosure 1 (2010ad), DOE stated it would update SAR Section 2.2.1.4.1.1.2.4.2 to show that the bias used in establishing loading limits for

DOE spent nuclear fuel canisters conservatively envelopes any uncertainty associated with the limited availability of applicable benchmarks.

An important consideration in the development of critical limits is the determination and implementation of an administrative margin. As described in SAR Section 2.2.1.4.1.1.2.4.2, the administrative margin is an arbitrary margin ensuring subcriticality and turning the criticality limit function into an upper subcritical limit function. DOE described that this term is not applicable for use in postclosure analyses because there is no risk associated with a subcritical event. DOE described that in contrast to "traditional" nuclear criticality safety analyses and associated governing regulations, in which the purpose is to ensure prevention of criticality and corresponding protection of personnel and facilities, the purpose of the postclosure criticality evaluation is to determine the probability of a criticality event in the postclosure time period. The probability of criticality is then compared to a probability of 10^{-4} in 10,000 years to reach a decision relative to the inclusion or exclusion of a criticality event in the evaluation of the total system performance for the facility. Thus, DOE used a zero administrative margin and provided additional justification for its use of a zero administrative margin in DOE Enclosure (2009bv).

NRC Staff's Review of Criticality Code Validation

The NRC staff reviewed DOE's documentation on criticality models, calculations, and results. The NRC staff also performed calculations (Sippel, 2010aa) for a sample of the configurations DOE considered subcritical and confirmed that the resulting k_{eff} is below the upper subcritical limit. This NRC staff's review and the confirmatory analysis provides the NRC staff's confidence that DOE appropriately modeled the potentially critical configurations when determining reactivity. DOE's calculation of k_{eff} is reasonable because DOE modeled neutron physics using standard industry computer codes that the NRC staff previously approved for criticality computations, as described in NRC Section 8.4.4.1 (2000ab).

The upper subcritical limit was calculated properly and the calculation was performed in accord with the NRC staff-endorsed methods (NRC, 2005ac). The verification and validation methodology in Radulescu, et al. (2007aa) is reasonable to the NRC staff because the predictor variables used are considered sufficient to fully characterize the bias across the range of applicability. The NRC staff also notes that DOE's validation methodology and selection of benchmarks for DOE spent nuclear fuel were inadequate and inconsistent with DOE's updated methodology in Radulescu, et al. (2007aa). However, the statement in DOE Enclosure 1 (2010ad) that DOE would show that the bias used in establishing loading limits for DOE spent nuclear fuel canisters conservatively envelopes any uncertainty associated with the limited availability of applicable benchmarks and the reasonable validation of the criticality model for commercial spent nuclear fuel provides NRC staff confidence that the DOE conclusion to exclude criticality from the performance assessment is reasonable.

The NRC staff reviewed DOE's justification for an administrative margin of subcriticality of zero. In a traditional criticality analysis, the administrative margin is used to protect against the possibility that the critical limit has been incorrectly defined. DOE considered code validation and conservatism in modeling parameters when assessing the proposed administrative margin of zero. In criticality calculations, some minimum level of assurance is sought to determine that the evaluated conditions are subcritical. In DOE Enclosure 15 (2009aj), DOE discussed the estimated margin in DOE spent nuclear fuel, and NRC staff notes that there is significant margin associated with not taking BWR burnup credit for most assemblies. DOE discussed in DOE Enclosure 2 (2010ad) how conservative parameters, rather than mean as recommended in NUREG–1804, were used in calculating the probability of breach. Had mean parameters been used, it would have significantly decreased the probability of criticality reported in SAR Table 2.2-8. Therefore, the analysis of the probability of criticality is reasonable without the use of an administrative margin.

FEPs 2.1.14.15.0A Through 2.1.14.26.0A and 2.2.14.09.0A Through 2.2.14.12.0A, Summary of NRC Staff's Review for Criticality Features, Events, and Processes

The NRC staff reviewed the models, calculations, and results DOE used for screening out criticality at the proposed Yucca Mountain repository using risk-informed, performance-based review methods described in the YMRP. Important areas of the review included reviewing the inputs to and validation of the isotopic irradiation and depletion model, the criticality models, the probability models, and the methodologies used in the models. On the basis of the results of this review and on DOE's analyses in DOE Enclosure 10 (2009bv), DOE Enclosure 2 (2009bx), and DOE Enclosure 1 (2010ad), DOE's technical basis for excluding the criticality features, events, and processes (FEPs 2.1.14.15.0A through 2.1.14.26.0A and 2.2.14.09.0A through 2.2.14.12.0A) from the performance assessment on the basis of low probability is reasonable. The NRC staff notes that DOE developed an adequate technical basis for screening out the criticality event class on the basis of low probability because the overall probability of criticality (4.4×10^{-5}; SAR Table 2.2-8) is less than 10^{-4} in 10,000 years.

Summary of Technical Evaluation of Screening of the List of Features, Events, and Processes

As described previously, the NRC staff makes the following evaluation. First, DOE has identified all features, events, and processes related to either the geologic setting or to the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) that have been excluded. Second, DOE has provided a justification for exclusion of each feature, event, and process and DOE's criteria for exclusion (i.e., justification) on the basis of low probability, low consequence, or exclusion by regulation are reasonable. Third, DOE has provided an adequate technical basis for each feature, event, and process, excluded from the performance assessment, to support the conclusion that either the feature, event, or process is specifically excluded by regulation; the probability of the feature, event, and process falls below a probability of 10^{-4} in 10,000 years; or omission of the feature, event, and process does not significantly change the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment. Fourth, the technical basis for FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, and DOE's supplemental analyses, comprise an analysis of seismic effects on the elevation of the water table beyond the 10,000-year postdisposal period throughout the period of geologic stability. DOE reasonably excluded features, events, and processes from the performance assessments beyond the 10,000-year postdisposal period throughout the period of geologic stability and properly identified particular features, events, and processes to be included in these performance assessments.

2.2.1.2.1.3.3 Technical Evaluation for Formation of Scenario Classes

Summary of the DOE Approach to Formation of Scenario Classes

DOE described the approach to the definition of event class and scenario class formation in SAR Section 2.2.1.3. DOE defined an event class as consisting of all possible initiating events that are caused by a common natural process. DOE extended the definition to allow event classes to represent the aggregation of initiating events with a common characteristic, either natural or man-made. DOE stated that the objective of scenario class development is to define a limited set of scenario classes that could be analyzed quantitatively while still covering the range of possible future states of the repository system (SAR Section 2.2.1.3, p. 2.2-22).

SAR Section 2.2.1.3.1 presented the scenario class formation for (i) individual protection and (ii) groundwater protection. On the basis of the probabilities in SAR Section 2.2.2 (which were evaluated in the TER Section 2.2.1.2.2), DOE retained the following events—seismic, igneous, and early waste package and drip shield failure—and developed general scenario classes from these retained events. The DOE defined the nominal scenario as the scenario that does not include any of these events, but that accounts for all other included features, events, and processes. DOE stated that the broad event classes (seismic, igneous, and early waste package and drip shield failure) are independent but not mutually exclusive. To define mutually exclusive classes that encompass a complete span of possible future states of the repository system, DOE considered a total of eight independent combinations from the three events and the nominal scenario, and summarized those combinations in the diagram in SAR Figure 2.2-3. DOE concluded that these eight scenario class combinations cover all of the possibilities of damaging events or processes that could affect a waste package during the timeframe of the analysis (e.g., a waste package could be damaged by both an early failure event and a seismic event). DOE did not explicitly implement these eight mutually exclusive scenario classes and their probabilities for the computation of aggregated consequence estimates. DOE introduced simplifications (SAR Section 2.4.2.1) and derived consequence estimates on the basis of the three broad scenario classes (seismic, igneous, and early waste package and drip shield failure), the nominal scenario class, and their probabilities.

DOE discussed scenario class formation for the human protection standard in SAR Section 2.2.1.3.2 and defined assumptions to support the development of the scenario.

NRC Staff's Evaluation of Formation of Scenario Classes

The NRC staff evaluated the formation of scenario classes for performance assessments for individual protection and groundwater protection by evaluating whether they cover the full range of potential future states of the repository system. The NRC staff notes the classifications are adequate and comprehensive as they encompass all possibilities (aside from human intrusion) leading to radionuclide release consistent with the set of included features, events, and processes. These four scenarios are not mutually exclusive (e.g., an initially failed waste package could also be affected by a seismic event). However, to assess the effect of combined scenarios, DOE considered the total of eight possible combinations from the three event and the nominal scenarios. These eight scenario class combinations cover all of the possibilities of damaging events or processes that could affect a waste package during the timeframe of the analysis. Therefore, DOE left no situation that could lead to radionuclide release without consideration and evaluation. DOE considered the effect of these eight combinations in determining aggregated dose estimates.

DOE's formation of scenario classes is reasonable on the following basis. DOE developed an initial set of scenario classes that covers all possible included degradation processes and events that could lead to release of radionuclides. From these broad scenario classes, DOE considered mutually exclusive and complete class combinations, and explained how these mutually exclusive class combinations were accounted for in the aggregated consequence estimates in the performance assessment computations. DOE clearly documented the set of scenario classes. The NRC staff notes that the DOE-defined scenario classes to support its performance assessments are complete (i.e., the classes account for all of the included features, events, and processes potentially leading to release of radionuclides) and that no relevant event was overlooked. Thus, DOE provided a technically sound basis for the formation of scenario classes. The scenario classes—nominal, seismic, igneous, and early waste package and drip shield failure—as defined by DOE, are reasonable classes to support a scenario screening process. The NRC staff notes that DOE considered class combinations that cover all possibilities leading to radionuclide release consistent with the set of included features, events, and processes and that DOE assessed the effect of these combinations in the dose estimates (SAR Section 2.4.2.1). Thus, the scenario classes for performance assessments for the individual protection and groundwater protection are complete, clearly documented, and technically supported.

The NRC staff notes the DOE analysis for human intrusion is reasonable, on the basis of the NRC staff's evaluation of the human intrusion scenario analysis performed in TER Section 2.2.1.4.2.3.

2.2.1.2.1.3.4 Screening of Scenario Classes

Summary of the DOE Approach to Screening of Scenario Classes

In SAR Section 2.2.1.4, DOE described step 4 (screening for scenario classes) of the scenario analysis (SAR Section 2.2.1). DOE included the following four scenario classes for the performance assessment computations to analyze individual protection: nominal, early failure, seismic, and igneous scenario classes. DOE asserted that all of these scenario classes have a probability greater than 10^{-4} in 10,000 years.

For the performance assessment to analyze groundwater protection, DOE excluded the igneous scenario class on the basis that its probability is less than 0.1 in 10,000 years.

The human intrusion scenario class is excluded from DOE's performance assessments to analyze individual protection and groundwater protection.

NRC Staff's Evaluation of Screening of Scenario Classes

DOE screened scenario classes on the basis of probability, consequences, and consistency with regulations. Accordingly, the NRC staff's review of DOE's exclusion of scenarios classes on the basis of low probability used the results of the NRC staff's evaluation documented in TER Section 2.2.1.2.2. In addition, the NRC staff reviewed whether the scenario classes that DOE ruled out by regulation were identified and justified.

In SAR Section 2.2.1.3, DOE defined scenario classes referred to as nominal, early failure, seismic, igneous scenario, and human intrusion, which were used as the starting point for the screening of scenario classes. As described in TER Section 2.2.1.2.1.3.3, these scenario classes are reasonable because the scenario classes were clearly documented and technically justified.

In SAR Section 2.2.1.3.1, DOE stated that, on the basis of probabilities described in SAR Section 2.2.2, seismic, igneous, and early waste package and drip shield failure were retained in the formation of scenario classes used in the performance assessments. The NRC staff reviewed the DOE basis for estimating these event probabilities in TER Section 2.2.1.2.2.3 and noted that the event probability for each of these events exceeds 1 chance in 10,000 of occurring within 10,000 years of disposal. The NRC staff also noted that DOE appropriately considered information from site and regional characterization, natural analog studies, and repository design in its evaluation of probability for each of the events.

DOE included the nominal, early failure, seismic, and igneous scenarios in the performance assessment computations to analyze individual protection. The human intrusion scenario was analyzed separately. The NRC staff notes that the inclusion of the four scenario classes is well supported. As stated in the previous section on formation of scenario classes, the four scenario classes incorporate all events (human intrusion aside) that could lead to significant radionuclide release. DOE's exclusion of human intrusion from the performance assessment is reasonable.

DOE included the nominal, early failure, and seismic scenarios in the performance assessment to analyze the groundwater protection scenario and excluded the igneous and human intrusion scenarios. The exclusion of the igneous scenario on the basis that igneous events are unlikely (i.e., a chance less than 1 in 100,000 per year of occurring) is reasonable on the basis of the NRC staff's evaluation in TER Section 2.2.1.2.2. In that section, staff noted that it was reasonable to consider the probability of igneous events to be below 1 in 100,000 per year of occurring; thus, DOE provided sufficient technical basis to exclude the igneous scenario. The igneous scenario class is sufficiently broad to encompass a range of events. Thus the NRC staff noted that this scenario class was not prematurely excluded by a narrow definition. Finally, exclusion of human intrusion from the groundwater protection scenario is reasonable, because such a scenario is to include only undisturbed performance.

DOE's screening of scenario classes is reasonable on the basis of the previously described evaluations. DOE's screening of scenario class was comprehensive, clearly documented, and technically sound. Scenario classes that DOE excluded from the performance assessment on the basis that they are specifically ruled out by the regulation were identified, and sufficient justifications were provided. Those scenario classes that DOE screened from the performance assessment on the basis that their probabilities fall below 10^{-4} in 10,000 years were identified and justified.

DOE included nominal, early failure, seismic, and igneous scenario classes in the performance assessment for individual protection, and this is reasonable on the basis that the probabilities of these scenario classes exceed 10^{-8} per year (probabilities evaluated in TER Section 2.2.1.2.2) and that these scenario classes incorporate all included events that could lead to significant radionuclide release. For the groundwater protection scenario, DOE excluded the igneous scenario class from the performance assessment model, and this exclusion is reasonable because DOE provided sufficient justification to show that igneous events are unlikely. In the individual protection and groundwater protection computations, DOE excluded the human intrusion scenario from the performance assessments, and this exclusion is reasonable because DOE provided sufficient justification. DOE's screening of scenario classes is reasonable for performance assessments to analyze individual protection and groundwater protection. Evaluation of event classes in regard to computations to analyze human intrusion is included in TER Section 2.2.1.4.2.

The NRC staff has reviewed the information in the SAR and other DOE information and notes that the scenario classes that have been screened out from the performance assessment calculations were justified.

2.2.1.2.1.4 NRC Staff Conclusions

NRC staff notes that the DOE approach for Scenario Analysis is consistent with the guidance in the YMRP. The NRC staff also notes that the DOE technical approach for scenario analysis discussed in this chapter is reasonable to support performance assessments.

2.2.1.2.1.5 References

Aiyanger, A.K., B.W. Neuberger, P.G. Oberson, and S. Ankem. 2005aa. “The Effect of Stress Level and Grain Size on the Ambient Temperature Creep Deformation Behavior of an Alpha Ti-1.6 Wt Pct V Alloy.” *Metallurgical and Materials Transactions A*. Vol. 36A. pp. 637–644.

American Nuclear Society. 1983aa. "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." ANSI/ANS–8.1–1983. La Grange Park, Illinois: American Nuclear Society.

American Society of Mechanical Engineers. 2001aa. *ASME Boiler and Pressure Vessel Code*. New York City, New York: American Society of Mechanical Engineers.

Ankem, S., C.A. Greene, and S. Singh. 1994aa. “Time Dependent Twinning During Ambient Temperature Creep of Alpha-Ti-Mn Alloys.” *Scripta Metallurgica*. Vol. 30. pp. 803–808.

AREVA 2004. "Reactor Record Uncertainty Determination." 32-5041666-02, Rev 02. ACC: DOC. 20040623.0002.ML092310715. ML092310716. 5/26/04.

ASM International. 2003aa. *ASM Handbook: Heat Treating.* Vol. 4. Metal Park, Ohio: ASM International.

ASM International. 1994aa. *Materials Properties Handbook: Titanium Alloys*. Metal Park, Ohio: ASM International.

ASTM International. 2004aa. “Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper, Low-Carbon Nickel-Chromium-Molybdenum-Tantalum, and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip.” B 475-04: Annual Book of ASTM Standards. Vol. 02.04. West Conshohocken, Pennsylvania: ASTM International.

Beavers, J.A. and C.L. Durr. 1991aa. NUREG/CR–5598, “Immersion Studies on Candidate Container Alloys for the Tuff Repository.” Washington, DC: NRC.

Beavers, J.A., N.G. Thompson, and C.L. Durr. 1992aa. NUREG/CR–5709, “Pitting, Galvanic, and Long-Term Corrosion Studies on Candidate Container Alloys for the Tuff Repository.” Washington, DC: NRC.

Brossia, C.S. and G.A. Cragnolino. 2000aa. “Effects of Environmental, Electrochemical, and Metallurgical Variables on the Passive and Localized Dissolution of Ti Grate 7.” CORROSION 2000. Paper No. 00211. Houston, Texas: NACE International.

BSC. 2007be. “Subsurface Geotechnical Parameters Report.” ANL–SSD–GE–000001. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2006aa. "Analysis of Alcove 8/Niche 3 Flow and Transport Tests." ANL–NBS–HS–000056. Rev. 00. ACN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005aa. "Analysis of Dust Deliquescence for FEP Screening." ANL–EBS–MD–000074. Rev. 01. AD 01, ACN 001, ACN 002. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005an. "Creep Deformation of the Drip Shield." CAL–WIS–AC–000004. Rev. 00A. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ab. "Aging and Phase Stability of Waste Package Outer Barrier." ANL–EBS–MD–000002. Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ae. "Aqueous Corrosion Rates for Waste Package Materials." ANL–DSD–MD–000001. Rev. 01. ACN 001, ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004al. "Drift Degradation Analysis." ANL–EBS–MD–000027. Rev. 03. ACN 001, ACN 002, ACN 003, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004aq. "Evaluation of Potential Impacts of Microbial Activity on Drift Chemistry." ANL–EBS–MD–000038. Rev. 01. ACN 01, ACN 02, ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004as. "General Corrosion and Localized Corrosion of the Drip Shield." ANL–EBS–MD–000004. Rev. 02. AD 01, ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004be. "Seepage Model for PA Including Drift Collapse." MDL–NBS–HS–000002. Rev. 3. ACN 001. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bi. "Yucca Mountain Site Description." TDR–CRW–GS–000001. Rev. 02 ICN 01. ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Cao, T. 2010aa. "Yucca Mountain Confirmatory Drift Degradation Calculations." Memo (September 21) T. Cao to J. Guttmann (NRC). ML102640197. Washington, DC: NRC

CNWRA and NRC. 2008aa. "Risk Insights Derived From Analyses of Model Updates in the Total-system Performance Assessment Version 5.1 Code." ML082240343. San Antonio, Texas: CNWRA.

Dassault Systèmes Simulia Corp. 2009aa. "Abaqus V6.9-1." Providence, Rhode Island: Dassault Systèmes Simulia Corp.

DOE. 2010ad. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (January 20) J.R. Williams to J.H. Sulima (NRC). ML100210164. Washington, DC: DOE, Office of Technical Management.

DOE. 2010ah. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (January 7) J.R. Williams to J.H. Sulima (NRC). ML100110028. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ab. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550101. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ae. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 1." Letter (January 23) J.R. Williams to J.H. Sulima (NRC). ML090260710. Washington, DC: DOE, Office of Technical Management.

DOE. 2009af. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 3." Letter (March 4) J.R. Williams to J.H. Sulima (NRC). ML091830594. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ah. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (March 3) J.R. Williams to J.H. Sulima (NRC). ML090860902. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ai. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report, Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (March 23) J.R. Williams to J.H. Sulima (NRC). ML090830357. Washington, DC: DOE, Office of Technical Management.

DOE. 2009aj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report, Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (March 24) J.R. Williams to J.H. Sulima (NRC). ML090840280. Washington, DC: DOE, Office of Technical Management.

DOE. 2009al. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (March 26) J.R. Williams to J.H. Sulima (NRC). ML090860424. Washington, DC: DOE, Office of Technical Management.

DOE. 2009av. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bo. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (June 1) J.R. Williams to J.H. Sulima (NRC). ML091530403. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bv. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (March 25) J.R. Williams to J.H. Sulima (NRC). ML090900069. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bw. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (October 26) J.R. Williams to J.H. Sulima (NRC). ML093070154. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bx. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (December 15) J.R. Williams to J.H. Sulima (NRC). ML093500245. Washington, DC: DOE, Office of Technical Management.

DOE. 2009by. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (June 4) J.R. Williams to J.H. Sulima (NRC). ML091560538. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bz. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (June 25) J.R. Williams to J.H. Sulima (NRC). ML091760913. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ca. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (July 31) J.R. Williams to J.H. Sulima (NRC). ML092150623. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cb. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (June 5) J.R. Williams to J.H. Sulima (NRC). ML091590581. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cc. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (August 12) J.R. Williams to J.H. Sulima (NRC). ML092250006. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cd. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 6." Letter (November 5) J.R. Williams to J.H. Sulima (NRC). ML093090335. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ce. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 6." Letter (November 17) J.R. Williams to J.H. Sulima (NRC). ML093220119. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cf. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 6." Letter (November 24) J.R. Williams to J.H. Sulima (NRC). ML093360134. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cg. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 6." Letter (December 3) J.R. Williams to J.H. Sulima (NRC). ML093380138. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ch. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 6." Letter (December 4) J.R. Williams to J.H. Sulima (NRC). ML093410044. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ci. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 7." Letter (October 30) J.R. Williams to J.H. Sulima (NRC). ML093060134. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 3." Letter (May 7) J.R. Williams to J.H. Sulima (NRC). ML091280184. ML091280185. Washington, DC: DOE, Office of Technical Management.

DOE. 2009co. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (August 11) J.R. Williams to J.H. Sulima (NRC). ML092310639. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cq. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 3." Letter (December 4) J.R. Williams to J.H. Sulima (NRC.) ML093410044. Washington, DC: DOE, Office of Technical Management.

DOE. 2009gp. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (June 19) J.H. Sulima to J.R. Williams (NRC). ML091730195. Washington, DC: DOE, Office of Technical Management.

DOE. 2009gq. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 4." Letter (July 1) J.H. Sulima to J.R. Williams (NRC). ML091880036. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2004aa. "Criticality Model." CAL–DS0–NU–000003. Rev. 00A. ECN 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2004ab. "Isotopic Model for Commercial SNF Burnup Credit." CAL–DSU–NU–000007. Rev. 00B. ECN 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2003aa. "Commercial Spent Nuclear Fuel Waste Package Misload Analysis." CAL–WHS–MD–000003. Rev. 00A. ECN 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Drefahl, K., P. Wincierz, U. Zwicker, and P. Delarbre. 1985aa. "The 230,000 h Creep Properties of Titanium Produced From Electrolytic and Sponge Material and TiAl6V4 Alloy at 20 °C." Proceedings from the 5th International Conference on Titanium, Congress-Center, Munich, Germany. pp. 2,387–2,394.

Dunn, D.S., Y.-M. Pan, L. Yang, and G.A. Cragnolino. 2006aa. "Localized Corrosion Susceptibility of Alloy 22 in Chloride Solutions: Part 2—Effect of Fabrication Processes." *Corrosion*. Vol. 62, No. 1. pp. 3–12.

Fix, D.V., J.C. Estill, L.L. Wong, and R.B. Rebak. 2004aa. "General and Localized Corrosion of Austenitic and Borated Stainless Steels in Simulated Concentrated Ground Waters." Proceedings of the 2004 ASME/JSME Pressure Vessel and Piping Conference: Transportation, Storage, and Disposal of Radioactive Materials. Report No. PVP–Vol. 483. San Diego, California.

Gauthier, J.H., M.L. Wilson, D.J. Borns, and B.W. Arnold. 1996aa. "Impacts of Seismic Activity on Long-Term Repository Yucca Mountain." Proceedings of the American Nuclear Society Topical Meeting on Methods of Seismic Hazards Evaluation: Focus '95, September 18–20, 1995, Las Vegas, Nevada. pp. 159–168.

Glass, R.S., G.E. Overturf, R.E. Garrison, and R.D. McCright. 1984aa. "Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff." Report UCID–20174. Livermore, California: Lawrence Livermore National Laboratory.

Green, R.T., C. Manepally, R.W. Fedors, and M.M. Roberts. 2008aa. "Examination of Thermal Refluxing in *In-Situ* Heater Tests." San Antonio, Texas: CNWRA

He, X, D.S. Dunn, and A.A. Csontos. 2007ab. "Corrosion of Similar and Dissimilar Metal Crevices in the Engineered Barrier System of a Potential Nuclear Waste Repository." *Electrochimica Acta*. Vol. 52. pp. 7,556–7,569.

Itasca International, Inc. 2004aa. "Fast Lagrangian Analysis of Continua (FLAC®)." Minneapolis, Minnesota: Itasca International, Inc.

Itasca International, Inc. 2004ab. "Particle Flow Code in Two Dimensions (PFC2D™)." Minneapolis, Minnesota: Itasca International, Inc.

Itasca International, Inc. 2004ac. "Universal Distinct Element Code (UDEC™)." Minneapolis, Minnesota: Itasca International, Inc.

Jaeger, J.C., N.G.W. Cook, and R.W. Zimmerman. 2007aa. *Fundamentals of Rock Mechanics*. 4th Edition. Malden, Massachusetts: Blackwell Publishing.

Kaiser, P.K., M.S. Diederichs, C.D. Martin, J. Sharp, and W. Steiner. 2000aa. “Underground Works in Hard Rock Tunnelling and Mining.” Proceeding of GeoEng 2000, Melbourne, Australia. Lancaster, Pennsylvania: Technomic Publishing Company. pp. 841–926.

Kemeny J.M. and N.G.W. Cook. 1992aa. “Water Table Change Due to a Normal Faulting Earthquake.” *Section 6: Demonstration of a Risk-Based Approach to High-Level Waste Repository Evaluation—Phase 2*. EPRI–TR–100384. Palo Alto, California: Electric Power Research Institute.

Kiessel, W.R. and M.J. Sinnott. 1953aa. “Creep Properties of Commercially Pure Titanium.” *Journal of Metals*. Vol. 5. pp. 331–337.

Lawrence Livermore National Laboratory. 1998aa. “Nonisothermal Unsaturated Flow and Transport (NUFT).” Livermore, California: Lawrence Livermore National Laboratory.

Leslie, B. 2010aa. "Table Summarizing Review of all Features, Events, and Processes." Memo (December 2) to J. Guttmann (NRC). ML102720603. Washington, DC: NRC.

Lin C., B. Leslie, R. Codell, H. Arlt, and T. Ahn. 2003aa. “Potential Importance of Fluoride to Performance of the Drip Shield.” Proceedings of the American Nuclear Society 10th International High-Level Radioactive Waste Management Conference, March 30–April 2, 2003, Las Vegas, Nevada.

Lister, T., R. Mizia, A. Erickson, and S. Birk. 2007aa. “Electrochemical Corrosion Testing of Borated Stainless Steel Alloys.” INL/EXT–07–12633. Rev. 1. Idaho Falls, Idaho: Idaho National Laboratory.

Lister T., R. Mizia, A. Erickson, and T. Trowbridge. 2007ab. “Electrochemical Corrosion Testing of Neutron Absorber Materials.” INL/EXT–06–11772. Rev. 1. Idaho Falls, Idaho: Idaho National Laboratory.

Liu, Z. and G. Welsch. 1988aa. “Literature Survey on Diffusivities of Oxygen, Aluminum, and Vanadium in Alpha Titanium, Beta Titanium, and Rutile.” *Metallurgical Transactions A*. Vol. 19A, No. 4. pp.121–125.

Liu, Z. and G. Welsch. 1988ab. “Effects of Oxygen and Heat Treatment on the Mechanical Properties of Alpha and Beta Titanium Alloys.” *Metallurgical Transactions A*. Vol. 19a, No. 3. pp. 527–542.

Manepally, C., A. Sun, R. Fedors, and D. Farrell. 2004aa. “Drift-Scale Thermohydrological Process Modeling—In-Drift Heat Transfer and Drift Degradation.” CNWRA 2004-05. ML042160447. San Antonio, Texas: CNWRA

McCright, R.D., W.G. Halsey, and R.A. Van Konynenburg. 1987aa. “Progress Report on the Results of Testing Advanced Conceptual Design Metal Barrier Materials Under Relevant Environmental Conditions for a Tuff Repository.” Report No. UCID–21044. Livermore, California: Lawrence Livermore National Laboratory.

Miller, W.H., R.T. Chem, and E.A. Starke. 1987aa. “Microstructure, Creep and Tensile Deformation in Ti-6Al-2Nb-1Ta-0.8Mo.” *Metallurgical Transactions A*. Vol. 18a. pp. 1,451–1,468.

Mintz, T. and X. He. 2009aa. "Modeling of Hydrogen Uphill Diffusion in Dissimilar Titanium Welds." Proceedings of the CORROSION 2009 Conference. Paper No. 09430. Houston, Texas: NACE International.

Moskalenko, V.A. and V.A. Puptsova. 1972aa. "Thermal Activation Parameters of the Plastic Deformation of Single Crystals of α-Titanium." *Fizika Metallov I Metallovedenie*. Vol. 34. pp. 1,264–1,269.

National Research Council. 1995aa. *Technical Bases for Yucca Mountain Standards.* Washington, DC: National Academies Press.

National Research Council. 1992aa. *Ground Water at Yucca Mountain: How High Can It Rise?* Washington, DC: National Academies Press.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2005ac. Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuel and Material Facilities." Rev. 1. Washington, DC: NRC.

NRC. 2005af. "Implementation of a Dose Standard After 10,000 Years: Proposed Rule." *Federal Register*. Vol. 70, No. 173. pp. 53313–53320. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC. 2000ab. NUREG–1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." Washington, DC: NRC, Spent Fuel Project Office.

NRC. 1998aa. Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities. Rev. 0. Washington, DC: NRC.

Nuclear Energy Agency. 2006aa. "The NEA International FEP Database, Version 2.1." Issy-les-Moulineaux, France: Organisation for Economic Cooperation and Development, Nuclear Energy Agency.

Nuclear Energy Agency. 2001aa. "Scenario Development Methods and Practice: An Evaluation Based on the NEA Workshop on Scenario Development, Madrid, May 1999." Paris, France: Organisation for Economic cooperation and Development, Nuclear Energy Agency.

Nuclear Energy Agency. 2000aa. "Features, Events, and Processes (FEPs) for Geologic Disposal of Radioactive Waste: An International Database." Paris France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Nuclear Energy Agency. 1997ab. “Safety Assessment of Radioactive Waste Repositories: An International Database of Features, Events, and Processes.” Ver. 3. SAM–J012–R1. Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Odegard, B.C. and A.W. Thompson. 1974aa. “Low Temperature Creep of Ti-6Al-4V.” *Metallurgical Transactions A*. Vol. 5A. pp. 1,207–1,213.

Ofoegbu, G., R. Fedors, C. Grossman, S. Hsiung, L. Ibarra, C. Manepally, J. Myers, M. Nataraja, O. Pensado, K. Smart, and D. Wyrick. 2007aa. “Summary of Current Understanding of Drift Degradation and Its Effects on Performance at a Potential Yucca Mountain Repository.” Rev. 1. CNWRA 2006-02. ML071030115. San Antonio, Texas: CNWRA.

Okada, T. 1983aa. “Factors Influencing the Cathodic Charging Efficiency of Hydrogen by Modified Titanium Electrodes.” *Electrochimica Acta*. Vol. 28. pp. 1,113–1,120.

Orava, R.N. 1967aa. “On the Deformation Mechanisms in Alpha-Titanium Below 0.4 T_M.” *Scripta Metallurgical*. Vol. 1. pp. 153–156.

Pabalan, R.T. 2010aa. “Quantity and Chemistry of Water Contacting Engineered Barriers Integrated Subissue.” Electronic Scientific Notebook 930E. San Antonio, Texas: CNWRA.

Pan, Y.-M., D.S. Dunn, and G.A. Cragnolino. 2005aa. “Topologically Close-Packed Phase Precipitation and Thermal Stability in Alloy 22.” *Metallurgical and Materials Transactions.* Vol. 36A. pp. 1,143–1,151.

Perfect, D.L., C.C. Faunt, W.C. Steinkampf, and A.K. Turner. 1995aa. “Hydrochemical Data Base for the Death Valley Region, California and Nevada.” USGS Open-File Report 94-305. Denver, Colorado: U.S. Geological Survey.

Pickett, D.A. and B.W. Leslie. 1999aa. “An Audit of the DOE Treatment of Features, Events, and Processes at Yucca Mountain, Nevada, With Emphasis on the Evolution of the Near-Field Environment.” San Antonio, Texas: CNWRA.

Radulescu, G., D.E. Mueller, S. Goluoglu, D.F. Hollenbach, and P.B. Fox. 2007aa. “Range of Applicability and Bias Determination for Postclosure Criticality of Commercial Spent Nuclear Fuel.” ACC: LLR.20071120.0179. ML092310677. Oak Ridge, Tennessee: Oak Ridge National Laboratory.

Rishel, D.M., J.D. Smee, and B.F. Kammenzind. 2000aa. “The Corrosion Behavior of Hafnium in High-Temperature Water Environments.” B–T–3232. West Mifflin, Pennsylvania: Bettis Atomic Power Laboratory.

Rogers, J.W., Jr., K.L. Erickson, D.N. Belton, R.W. Springer, T.N. Taylor, and J.G. Berry. 1988aa. “Low Temperature Diffusion of Oxygen in Titanium and Titanium Oxide Films.” *Applied Surface Science*. Vol. 35, No. 1. pp. 137–152.

Salve, R. and T.J. Kneafsey. 2005aa. “Vapor-Phase Transport in the Near-Drift Environmental at Yucca Mountain.” *Water Resources Research*. Vol. 41. W01012. Washington, DC: American Geophysical Union.

Sippel, T. 2010aa. “Yucca Mountain Confirmatory Criticality Calculations.” Memo (August 26) to J. Guttmann (NRC). ML102380107. Washington, DC: NRC.

Smailos, E. and R. Köster. 1987aa. “Corrosion Studies on Selected Packaging Materials for Disposal of High Level Wastes.” *Materials Reliability in the Back End of the Nuclear Fuel Cycle*. Proceedings of a Technical Committee Meeting, Vienna, Austria, September 2–5, 1986. TECHDOC–421. Vienna, Austria: International Atomic Energy Agency. pp. 7–24.

Smailos, E., W. Schwarzkopf, and R. Storch. 1992aa. “Corrosion Studies on Packaging Materials for High-Level Waste Dispsoal in a Rock-Salt Repository.” 12th Scandinavian Corrosion Congress & Eurocorr '92. Espoo, SF. May 31–June 4, 1992. Vol. II. pp. 327–338. Helsinki, Finland: Corrosion Society of Finland.

SNL. 2008aa. “CSNF Loading Curve Sensitivity Analysis.” ANL–EBS–NU–000010. Rev. 00. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ab. “Features, Events, and Processes for the Total System Performance Assessment: Analyses.” ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ac. “Features, Events, and Processes for the Total System Performance Assessment: Methods.” ANL–WIS–MD–000026. Rev. 00. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ad. “Postclosure Nuclear Safety Design Bases.” ANL–WIS–MD–000024. Rev. 01. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ae. “Screening Analysis of Criticality Features, Events, and Processes for License Application.” ANL–DS0–NU–00001. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. “Total System Performance Assessment Model/Analysis for the License Application.” MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008aj. “Multiscale Thermohydrologic Model.” ANL–EBS–MD–000049. Rev. 03. ADD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008al. “Waste Package Flooding Probability Evaluation.” CAL–DN0–NU–000002. Rev. 00C. AD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ab. “Atmospheric Dispersal and Deposition of Tephra From a Potential Volcanic Eruption at Yucca Mountain, Nevada.” MDL–MGR–GS–000002. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ai. “Drift-Scale THC Seepage Model.” MDL–NBS–HS–000001. Rev. 05. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ak. “Engineered Barrier System: Physical and Chemical Environment.” ANL–EBS–MD–000033. Rev. 06. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007al. "General Corrosion and Localized Corrosion of Waste Package Outer Barrier." ANL–EBS–MD–000003. Rev. 03. ACN 01, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007am. "Geochemistry Model Validation Report: Material Degradation and Release Model." ANL–EBS–GS–000001. Rev. 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ax. "Saturated Zone Site-Scale Flow Model." MDL–NBS–HS–000011. Rev. 03. ACN 01, ERD 01, ERD 02. ERD 03. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ba. "Site-Scale Saturated Zone Transport." MDL–NBS–HS–000010. Rev. 03. ACN 01, AD 001. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bb. "Stress Corrosion Cracking of Waste Package Outer Barrier and Drip Shield Materials." ANL–EBS–MD–000005. Rev. 04. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bf. "UZ Flow Models and Submodels." MDL–NBS–HS–000006. Rev. 03. ACN 01, ERD 01, ERD 02, ERD 03. Las Vegas, Nevada: Sandia National Laboratories.

Stetina, K. 1980aa. "Creep of Titanium Between 30 °C and 300 °C." Proceedings of the International Conference on Engineering Aspects of Creep, September 1980, University of Sheffield, England. Paper No. C194/80. London, England: Institute of Mechanical Engineers.

Stetina, K. 1969aa. "Comments on 'On the Deformation Mechanisms in Alpha-Titanium Below 0.4 T_M.'" *Scripta Metallurgica*. Vol. 3. pp. 57–58.

Stonestrom, D.A., D.E. Prudic, R.J. Laczniak, K.C. Akstin, R.A. Boyd, and K.K. Henkelman. 2003aa. "Estimates of Deep Percolation Beneath Native Vegetation, Irrigated Fields, and the Amargosa-River Channel, Amargosa Desert, Nye County, Nevada." USGS Open-Field Report 03-104. Denver, Colorado: U.S. Geological Survey.

Teper, B. 1991aa. "Activities Aimed at Qualification of a HLW Disposal Canister." Proceedings of the Second Annual International Conference on High-Level Radioactive Waste Management, Las Vegas, Nevada, April 1991. La Grange Park, Illinois: American Nuclear Society.

Thompson, A.W. and B.C. Odegard. 1973aa. "The Influence of Microstructure on Low Temperature Creep of Ti-5Al-2.5Sn." *Metallurgical Transactions*. Vol. 4. pp. 899–908.

Turner, D.R. and R.T. Pabalan. 1999aa. "Abstraction of Mechanistic Sorption Model Results for Performance Assessment Calculations at Yucca Mountain, Nevada." *Waste Management*. Vol. 19. pp. 375–388.

Waiting, D.J., J.A. Stamatakos, D.A. Ferrill, D.W. Sims, A.P. Morris, P.S. Justus, and K.I. Abou-Bakr. 2003aa. "Methodologies for the Evaluation of Faulting at Yucca Mountain, Nevada." Proceedings of the10th International High-Level Radioactive Waste Management Conference, Las Vegas, Nevada, March 30–April 2, 2003. La Grange Park, Illinois: American Nuclear Society. pp. 377–387.

Zeyfang, R., R. Martin, and H. Conrad. 1971aa. “Low Temperature Creep of Titanium.” *Materials Science and Engineering.* Vol. 8. pp. 134–140.

CHAPTER 3

2.2.1.2.2 Identification of Events With Probabilities Greater Than 10^{-8} Per Year

2.2.1.2.2.1 Introduction

This chapter evaluates the U.S. Department of Energy's (DOE) information on event probability used in its performance assessments evaluations. The U.S. Nuclear Regulatory Commission (NRC) staff evaluated information provided in the Safety Analysis Report (SAR) (DOE, 2008ab, 2009av), as supplemented by the DOE responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009aa–ad,aq,bd), and the references cited therein.

A performance assessment is a systematic analysis that answers the following triplet risk questions: What can happen? How likely is it to happen? What are the consequences? Scenario analysis answers the first question: What can happen? The NRC staff's evaluation of the DOE scenario analysis is documented in its Technical Evaluation Report (TER) Section 2.2.1.2.1. One result from the scenario analysis is the identification of the events to be included in the performance assessment calculations used to evaluate potential doses. This chapter addresses the second question: How likely is it that these events will happen?

A performance assessment evaluation considers events that have at least 1 chance in 100 million per year of occurring. DOE identified and described those events that exceeded this probability threshold. Performance assessments are also used to evaluate the human intrusion and groundwater protection calculations. These performance assessments have different considerations for event probabilities than those used for the individual protection calculation. The DOE approach for quantifying the event probabilities and the technical basis for determining the probability estimates assigned to each event type are addressed in this chapter.

2.2.1.2.2.2 Evaluation Criteria

This chapter of the TER documents the NRC staff's review of event probabilities used in its performance assessments evaluations, which is guided by the requirements of 10 CFR 63.114(a)(1), 10 CFR 63.114(a)(2), 10 CFR 63.114(a)(4), 10 CFR 63.114(a)(7), and 10 CFR 63.114(b).

10 CFR 63.114(a)(1) requires that any performance assessment used to evaluate repository performance objectives after permanent closure must include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region to the extent necessary, and information on the design of the engineered barrier system used to define, for 10,000 years after disposal, parameters and conceptual models used in the assessment.

10 CFR 63.114(a)(2) requires that any performance assessment must account for uncertainties and variabilities in parameter values, for 10,000 years after disposal, and provide for the technical basis for parameter ranges, probability distributions, or bounding values used in the performance assessment.

10 CFR 63.114(a)(4) requires that any performance assessment must consider only features, events, and processes consistent with the limits on performance assessment provided in 10 CFR 63.342. 10 CFR 63.342(a) states that a performance assessment need not consider very unlikely events (i.e., those that are estimated to have a less than 1 chance in 100 million per year of occurring). Further, 10 CFR 63.342(b) provides that the performance assessment used for the human intrusion and groundwater protection calculations excludes unlikely events (i.e., those events that are estimated to have an annual probability of occurring between 1 in 100,000 and 1 in 100 million).

10 CFR 63.114(a)(7) requires that the technical basis for models used to represent the 10,000 years after disposal in the performance assessment, such as comparisons made with outputs of detailed process-level models and/or empirical observations (e.g., laboratory testing, field investigations, and natural analogs), be provided.

10 CFR 63.114(b) states that the performance assessment methods used to address 10 CFR 63.114(a) are considered sufficient for the performance assessment for the period of time after 10,000 years and through the period of geologic stability. As defined in 10 CFR 63.302, the period of geologic stability means the time during which the variability of geologic characteristics and their future behavior in and around the Yucca Mountain site can be bounded (i.e., they can be projected within a reasonable range of possibilities). This period is defined to end at 1 million years after disposal. 10 CFR 63.342(c) requires that DOE's performance assessment project the continued effects of the features, events, and processes included in 10 CFR 63.342(a) beyond the 10,000-year postdisposal period through the period of geologic stability. 10 CFR 63.342(c)(1) specifies that DOE is to assess the effects of seismic and igneous activity scenarios, subject to the probability limits in 10 CFR 63.342(a) for very unlikely features, events, and processes, or sequences of events and processes.

The NRC staff used the review guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). YMRP Section 2.2.1.2.2 contains the following five acceptance criteria:

1. Events are adequately defined
2. Probability estimates for future events are supported by appropriate technical bases
3. Probability model support is adequate
4. Probability model parameters have been adequately established
5. Uncertainty in event probability is adequately evaluated

Additionally, YMRP Section 2.2.1 provides guidance to the NRC staff on a process to apply risk information in its review of the DOE SAR. Following the YMRP guidance, the NRC staff considered DOE's risk information (derived from DOE's treatment of multiple barriers) and risk insights that are identified in SAR Section 2.4.2.2.1.2. The level of detail of the NRC staff's review on particular parts of the SAR is based on the risk insights DOE developed and from consideration of the risk insights identified in NRC Appendix D (2005aa), as updated (CNWRA and NRC, 2008aa). Accordingly, the NRC staff used a graded approach in its review per the guidance identified in YMRP Section 2.2.1.

2.2.1.2.2.3 Technical Evaluation

In SAR Section 2.2.2, DOE considered events for inclusion into the postclosure performance assessments. Initial considerations included five event types: igneous events, seismic events, early failure events, criticality events, and human intrusion events. As described in TER Section 2.2.1.2.1, DOE screened out the individual criticality features, events, and processes and DOE determined that the probability of the nuclear criticality event class has less than 1 chance in 100 million per year of occurring.

DOE retained igneous activity, seismic activity, and early failure events in its performance assessment evaluation of the individual protection scenario. The NRC staff's evaluations of DOE information for igneous events, seismic events, and early failure events are documented in TER Sections 2.2.1.2.2.3.1, 2.2.1.2.2.3.2, and 2.2.1.2.2.3.3, respectively.

2.2.1.2.2.3.1 Technical Evaluation for Igneous Event Probabilities

This section presents the NRC staff's evaluation of information DOE presented to estimate the probability of future igneous events. The NRC staff reviewed SAR Sections 2.1 and 2.4; SAR Sections 1.1.6, 2.2.2.2, and 2.3.11; and material provided in response to the NRC staff's request for additional information (RAIs) (DOE, 2009aa,bd) and the references cited therein. DOE's description of past igneous activity in the Yucca Mountain region (SAR Sections 1.1.6 and 2.3.11) and the overall approach for treatment of igneous events in the SAR are summarized next.

DOE indicated that periods of igneous activity have resulted in the eruption of basalt magmas in the Yucca Mountain region during the last 11 million years and identified that the risk to the repository from future igneous activity could come from rising basalt magmas. The age and location of basaltic rocks that formed during at least six volcanic events in the last 5 million years, within approximately 50 km [31.1 mi] of the repository, are provided in SAR Figure 1.1-152. In presenting information on the location of volcanism in the Yucca Mountain region, DOE described the geologic and geophysical techniques it used to characterize past activities. DOE indicated that basalts in the Yucca Mountain region appear to be products of partial melting of lower lithospheric mantle material, but acknowledged that there is a poor understanding of the exact mechanism of mantle melting. DOE characterized the basaltic volcanism in the Crater Flat volcanic field, which is in close proximity to Yucca Mountain, as having a relatively long lifetime with a small volume of erupted material (SAR Section 1.1.6.1.1).

DOE included igneous events within the igneous event scenario class (SAR Section 2.2.1.3.1). DOE's Total System Performance Assessment (TSPA) evaluation divides the igneous event scenario class into separate modeling cases for intrusive events and extrusive events (SAR Section 2.4.1.2.3). Intrusive events involve the rise of molten rock (i.e., magma) from deep in the Earth with the magma intersecting the repository drifts (tunnels). DOE made the assumption that if magma flows into drifts, it damages the barrier capabilities of the drip shields and waste packages and allows subsequent radionuclide release through the hydrologic (water) transport pathway (SAR Section 2.3.11.3). DOE viewed extrusive events as an extension of intrusive events: after the magma has entered a repository drift and reached the surface, a conduit may develop from which most of the magma erupts, producing a volcano. Of the intrusive igneous events that intersect the repository footprint, DOE only considered a subset of these events to develop a conduit within the repository and form a volcano at the surface. DOE further assumed that only in some cases will waste packages be intersected by this conduit and release their contents into the rising magma. The magma and incorporated waste is then explosively erupted (expelled) from the surface volcano and transported by airborne dispersion for some distance downwind of the vent (SAR Section 2.3.11.4). Thus, the probabilities of intrusive and extrusive (volcanic eruptive) igneous events that may disperse waste and radionuclides into the environment, either in the subsurface or via atmospheric transport, are different (SAR Section 2.4.1.2.3). To assess the probability of a future igneous event intersecting the repository, DOE conducted a probabilistic volcanic hazard assessment (PVHA) using an elicitation process consisting of recognized experts (SAR Section 5.4).

Risk Perspective

If an igneous event occurred at the proposed repository, the igneous intrusion modeling case would constitute most of the calculated dose for the first 1,000 years following closure, as shown in SAR Figure 2.4-18(a), and is approximately half the calculated dose for the seismic ground motion modeling case in the ensuing 9,000 years. Moreover, DOE indicated that for the post-10,000-year period, the igneous intrusion modeling case and seismic ground motion modeling case provide approximately equal contributions to the total mean annual dose to the reasonably maximally exposed individual for the last 300,000 years of the time period. This statement (in SAR Section 2.1) is supported by the results presented in SAR Figure 2.4-18(b).

In SAR Section 2.4.2.2.1.2.3, DOE provided the probability-weighted consequences of igneous activity (intrusive and extrusive) using the probability distribution from the probabilistic volcanic hazard assessment review. For igneous intrusive events, DOE assumed that all waste packages will fail (approximately 11,000), whereas for an igneous extrusive event, DOE estimated only a few waste packages (<10) will result in radionuclide release (SAR Section 2.3.11.4.2.1). DOE identified that the probability-weighted igneous intrusive mean dose is estimated to be less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period and the median dose is less than 0.005 mSv/yr [0.5 mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.1). The DOE estimates for probability-weighted igneous extrusive (volcanic eruptive) mean dose are about 10^{-6} mSv/yr [0.0001 mrem/yr] for the 10,000-year period and for the median dose are less than 6×10^{-7} mSv/yr [6×10^{-5} mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.2). The difference in magnitude for the dose consequences between the two igneous scenarios (intrusive and extrusive) predominately results from the differing number of waste package failures estimated to occur for each scenario.

Summary of the DOE SAR on Igneous Event Probability

DOE evaluated the risk of future igneous activity, in part, by considering the probability that a future igneous event could intersect the repository. To quantify the probability of future igneous activity at the proposed Yucca Mountain repository, DOE conducted an expert elicitation review (probabilistic volcanic hazard assessment) (CRWMS M&O, 1996aa). This expert elicitation review resulted in a quantification of the mean annual probability of intersection of the repository by a future basaltic dike (an igneous intrusion) and its associated uncertainty distribution.

In SAR Sections 2.2.2.2.1–2.2.2.2.5, DOE described the probability of a future igneous event intersecting the repository, the technical basis for the probability estimate, the probability model support including alternative estimates of the intersection probability, the probability model parameters, and the uncertainties associated with the probability estimate, respectively. Subsequent to the probabilistic volcanic hazard assessment evaluation, DOE conducted an aeromagnetic survey and drilling program to increase confidence in site characterization results related to igneous activity. As described in TER Section 2.5.4, DOE updated the probabilistic volcanic hazard assessment evaluation (CRWMS M&O, 1996aa) in a study known as probabilistic volcanic hazard assessment-update (PVHA-U) (SNL, 2008ah). In SAR Section 5.4.1 (DOE, 2009av), DOE stated that the PVHA-U was conducted in a manner that is consistent with NUREG–1563 (NRC, 1996aa) and past practices. The average annual probability for an intrusive igneous event calculated in the PVHA-U is approximately twice as high as calculated in the original probabilistic volcanic hazard assessment evaluation (Boyle, 2008aa). DOE stated that the difference between these two event probability distributions would not significantly affect the estimates of repository performance over either 10,000 years or 1 million years, and in SAR Section 5.4.1, DOE further stated that the PVHA-U results are confirmatory of the original probabilistic volcanic hazard assessment technical basis (DOE, 2009av; also see Boyle, 2008aa).

NRC Staff's Evaluation of Igneous Event Probability

The NRC staff reviewed DOE's igneous event probability in SAR Sections 2.2.2.2.2, 2.2.2.2.4, and 2.2.2.2.5. These three SAR sections address the technical bases of probability estimates, probability model parameters, and uncertainty in event probability, respectively. The NRC staff's review focused on DOE's information that addressed the event definition (SAR Section 2.2.2.2.1) and probability model support(SAR Section 2.2.2.2.3). The NRC staff's evaluation of DOE's igneous intrusive event probability is provided next, following the NRC staff's evaluation of event definition and probability model support.

Event Definition

In SAR Section 2.2.2.2.1.1, DOE stated that the output of the probabilistic volcanic hazard assessment evaluation is the annual frequency of intersection of the proposed repository by an intrusive basaltic dike [CRWMS M&O, Section 4.2, Figure 4-32 (1996aa)]. The probabilistic volcanic hazard assessment expert elicitation program computed the mean annual probability of intersection of the proposed repository by an igneous basaltic dike as 1.5×10^{-8} per year (CRWMS M&O, 1996aa). The probabilistic volcanic hazard assessment results increased only slightly when these probability estimates were recalculated to reflect postelicitation changes to the size, shape, and location of the proposed repository footprint (BSC, 2004af); the recalculated mean annual probability from probabilistic volcanic hazard assessment thus became 1.7×10^{-8} per year (SAR Section 2.2.2.2.1.2). In the TSPA, DOE sampled a distribution of probability values for the likelihood of intrusive intersection with a mean value of 1.7×10^{-8} per year and computed the 5th and 95th percentiles of the uncertainty distribution at 7.4×10^{-10} and 5.5×10^{-8}, respectively.

DOE also calculated the proportion of the intersections that include development of a conduit (i.e., an eruption through the repository). It incorporated information from the probabilistic volcanic hazard assessment and model calculations that are supported by information obtained from studies of analog volcanoes with exposed intrusive rocks from depths of 200–300 m [656-984 ft] (SNL, 2007ae). DOE subsequently calculated that 28 percent of intrusive events would develop a volcanic conduit within the repository footprint (SAR Sections 2.2.2.2.1.3 and 2.3.11.4.2.1). As described in SAR Section 2.3.11.4.2.1.3, this fraction was determined by considering that a conduit can form at any location on a dike that intersects the repository and thus may not necessarily form within the footprint and by considering several other factors, including the distribution for the number of dikes in a swarm, consistent with analog basaltic volcanic events that included multiple dikes and in which conduit(s) formed on the widest dike. This conditional probability is for a conduit that develops within the repository. DOE then applied a second conditional probability of 0.297 (SAR Section 2.3.11.4.2.1) to represent the fraction of conduits that may intersect a drift containing waste packages and eject the waste contents through a volcanic vent (SAR Section 2.3.11.4.2.1).

Some PVHA panel members (CRWMS M&O, 1996aa) used event definitions for igneous events that included characteristics of both intrusive and extrusive events. For example, recurrence rates for intrusive events often were determined by interpreting the number of volcanic vents associated with a single event. However, the dike lengths used to represent these recurrence rates in probability models were independent of the relevant vent counts. See, for example, the discussion by McBirney in CRWMS M&O Appendix E (1996aa) that gave 90 percent weight to the 12-km [7.5-mi]-long chain of Quaternary Crater Flat volcanoes as representing a single event, but gave 90 percent weight that the dike supporting this event is less than 5 km [3.1 mi] {the dike must be at least 12 km [7.5 mi] long to feed the chain of volcanoes}. These inconsistencies in event definitions were resolved in the probabilistic volcanic hazard assessment-update evaluation (SNL, 2008ah), which used consistent definitions for intrusive and extrusive (volcanic) igneous event probabilities.

The probability of an intrusive disruption of the repository differed between probabilistic volcanic hazard assessment and its update by a factor of 1.8 (1.7×10^{-8} versus 3.1×10^{-8} per year, respectively). The event definition for the extrusive (volcanic) case in probabilistic volcanic hazard assessment-update was the formation of a conduit within the repository that would support an (explosive) eruption column; thus it was different and more specific than the probabilistic volcanic hazard assessment conditional probability of a conduit forming. The probability of an eruptive conduit event developing within the repository differed between probabilistic volcanic hazard assessment and its update by a factor of 2.5 (4.8×10^{-9} versus 1.2×10^{-8} per year, respectively). However, the two values are not directly comparable for the reason stated previously (Boyle, 2008aa). Therefore, any potential effects of inconsistencies related to event definitions within the probabilistic volcanic hazard assessment and the probabilistic volcanic hazard assessment-update evolutions have less than a factor of two effect on the uncertainty associated with the mean annual probability. Given the breadth of new information considered in the PHVA-U, the potential effect of uncertainty in event definitions would have less than a factor of two effect on the estimated mean annual igneous event probability. Because the mean doses DOE calculated for the intrusive and extrusive (volcanic) igneous cases (SAR Section 2.4) are on the order of 0.01 mSv/yr [1 mrem/yr] or less, the NRC staff notes that an approximate doubling of the igneous probabilities would have limited, if any, significance to risk (SNL, 2008ah; Boyle, 2008aa). Further information and review of the intrusive and extrusive (volcanic) igneous scenarios are provided by DOE in SAR Section 2.3.11 and by the NRC staff's evaluation in TER Section 2.2.1.3.10.

On the basis of the information provided in SAR Section 2.2.2.2 and DOE's statement that the updated probabilistic volcanic hazard assessment results confirmed the conclusions of the original probabilistic volcanic hazard assessment technical basis [SAR Section 5.4.1 (DOE, 2009av)], DOE has adequately defined an igneous intrusive event and an extrusive (volcanic eruptive) event that may affect the proposed repository. These definitions are consistent with their use in the performance assessment for the igneous abstraction. DOE has reasonably evaluated separate probabilities, including uncertainties, for the igneous intrusion and volcanic eruptive events.

Probability Model Support

In describing the geologic basis for the probabilistic volcanic hazard assessment evaluation (SAR Section 2.2.2.2.3.1), DOE indicated that the PVHA combined multiple alternative conceptual models into a single distribution that captured the uncertainty in the expert panel's conceptual models for the physical behavior of volcanism in the Yucca Mountain region. DOE also stated that for regional volcanism, no single conceptual model is appropriate because the underlying physical processes that control the precise timing and location of volcanic events within a particular region remain uncertain (BSC, 2004af). To support the probabilistic volcanic hazard assessment evaluation, DOE provided its elicitation panel with a variety of published information on igneous features, tectonics, and geophysical characteristics of the Yucca Mountain region (see CRWMS M&O, 1996aa). The probabilistic volcanic hazard assessment panel concluded that basaltic volcanism in the Yucca Mountain region resulted from complex interactions in the lithospheric mantle that produced episodes of small-volume basaltic magma. Because these mantle processes were viewed as uncertain, the probabilistic volcanic hazard assessment panel members did not explicitly include mantle processes in their probability models.

DOE has also indicated that past volcanic activity has occurred in the tectonically active Yucca Mountain region and could continue into the future with a very small probability of occurrence. During the period from 14 to 10 million years ago, major explosive eruptions involving rhyolite magma from volcanic centers lying roughly 20–40 km [12–25 mi] north of the repository site formed large caldera volcanoes and deposited the volcanic ash-flow tuffs of the region, some of which are the host rocks for the proposed repository. DOE concluded that the chance of a recurrence within the repository lifetime is less than 1 in 10,000 during 10,000 years [SAR Sections 2.2.2.2.2.1 and 2.3.11.2.1.1; see also Detournay, et al. (2003aa)] because there has been a sufficient time gap since caldera-forming volcanic activity ended about 8 million years ago (BSC, 2004bi). On the basis of available literature on the age and duration of this major phase of caldera-forming volcanism in the Yucca Mountain region (Sawyer, et al., 1994aa; Fleck, et al., 1996aa) and the NRC staff's consideration of the lifespans of caldera volcanoes in other parts of the world (e.g., Costa, 2008aa), DOE's conclusion has a reasonable technical basis.

Many probabilistic volcanic hazard assessment models for representing spatial variability in igneous events rely on interpretations of how tectonic processes (such as location of faults or distributions of crustal-scale stresses) affect the rise of magma to the Earth's surface. For example, many of the probabilistic volcanic hazard assessment source-zone models, described in CRWMS M&O Appendix E (1996aa), were defined by interpretations of tectonic influence on the spatial distribution of past events.

The NRC staff considers, in this regard, that a key concept discussed in BSC (2004af) is that although it is known that tectonic processes affect the rise of magma to the Earth's surface, a process-level understanding of tectonic controls on magma rise is beyond current scientific understanding. Thus, to address this conceptual uncertainty, probability models should consider a variety of alternative models for possible tectonic influences on the location of future igneous events at Yucca Mountain. The NRC staff notes that the probability models DOE provided considered an adequate variety of models over a range of scales to account for possible tectonic influences on spatial patterns of igneous events. These models include, at the broadest scale, the structural setting of the Yucca Mountain region within the southern Great Basin area of the Basin and Range Province, the more local influence of the Crater Flat structural domain adjacent to Yucca Mountain, and the presence of buried (igneous) anomalies (SAR Sections 2.2.2.2 and 2.3.11.2.1.1).

To support the probabilistic volcanic hazard assessment probability estimates developed prior to 2002, DOE characterized known basaltic igneous features within approximately 80.5 km [50 mi] of the Yucca Mountain site (BSC, 2004af). These investigations provided the location, age, and basic characteristics necessary to support probability estimates in the probabilistic volcanic hazard assessment evaluation. DOE also conducted geophysical investigations and borehole drilling to further characterize buried igneous features in the Yucca Mountain region (O'Leary, et al., 2002aa; Perry, et al., 2005aa). This new information was considered in the probabilistic volcanic hazard assessment-update evaluation (SNL, 2008ah). The NRC staff reviewed the information in these documents and determined that DOE provided support for the probability model development, by providing an appropriate level of detail on the location, age, and basic characteristics of igneous features in the Yucca Mountain region.

The NRC staff also conducted independent investigations in the Yucca Mountain region to support the evaluation of uncertainties in the location, age, and characteristics of buried igneous features (e.g., Magsino, et al., 1998aa; Stamatakos, et al., 1997ab; Hill and Connor, 2000aa; Hill and Stamatakos, 2002aa; Stamatakos, et al., 2007aa). Results from these investigations confirm that the location, age, and characteristics of buried igneous features were uncertain prior to 2002, but that these uncertainties were relatively small when considered in probability estimates after 2002. As shown by comparing the results of the probabilistic volcanic hazard assessment with its update, new information on igneous features has no more than a factor of two effect on the DOE probability estimate. Given the breadth of new information considered in the PHVA-U, the NRC staff notes that the potential effect of uncertainty in event definitions would have less than a factor of two effect on the estimated mean annual probability. DOE has sufficiently characterized igneous features in the Yucca Mountain region to support probability models for future igneous events that may affect the proposed repository at Yucca Mountain.

DOE discussed alternative estimates of the annual probability of an intrusive event intersecting the repository footprint (SAR Section 2.2.2.2.3.2). Both the NRC staff and the State of Nevada independently sponsored the development of these published models (SAR Table 2.2-18). Some of these models use methods and data developed after the 1996 probabilistic volcanic hazard assessment elicitation. Annual probability estimates for these published models range from 3×10^{-10} to 3×10^{-7}. DOE stated that these values cluster at slightly greater than 10^{-8} per year (SAR Section 2.2.2.2.3.2) and concluded that the apparent clustering near 10^{-8} per year provides confidence that the probabilistic volcanic hazard assessment probability estimate is robust. The NRC staff notes that the published probability model results do not strongly cluster at the value of the central tendency in the probabilistic volcanic hazard assessment probability distribution. However, the published values encompass a similar range to that of the PVHA and the published probability estimates for igneous events generally fall within the range of 10^{-8} to 10^{-7} per year. This range coincides with the probabilistic volcanic hazard assessment mean annual probability estimate.

In the discussion of probability model support (SAR Section 2.2.2.2.3.2), DOE did not address published models by Ho and Smith (1997aa) and Ho, et al. (2006aa). The NRC staff requested additional information from DOE to address these published probability estimates. In its response to the NRC staff, DOE (2009bd) stated that the calculations in Ho and Smith (1997aa) were performed as sensitivity analyses, which included parameter ranges selected from either expert knowledge or for "mathematical convenience," as stated in Ho and Smith, p. 621 (1997aa). DOE (2009bd) stated that the probability model approach developed in Ho and Smith (1997aa) was captured in the range of probability estimates Ho and Smith (1998aa) presented subsequently. The NRC staff reviewed the information in the DOE response and notes that the probability estimates in Ho and Smith (1998aa) reasonably represent the probability models developed in Ho and Smith (1997aa) using expert knowledge. Thus, DOE has appropriately considered the results from Ho and Smith (1997aa) in establishing confidence that the DOE probability estimate is robust. Although DOE stated that Ho, et al. (2006aa) did not present disruption probability results (DOE, 2009bd), a probability estimate of 10^{-7} is presented in Ho, et al., p. 121 (2006aa) on the basis of the author's interpretation of recurrence rates in Smith, et al.

(2002aa). The NRC staff notes that the probability estimate in Ho, et al. (2006aa) is consistent with the information DOE presented in SAR Section 2.2.2.2.3.2 and would not significantly affect the rationale DOE presented to support confidence in the DOE probability estimate.

As an independent confirmatory estimate, the NRC staff examined whether DOE's probability model results are consistent with past patterns of basaltic igneous events in the Yucca Mountain region that are younger than approximately 11 million years old in the Yucca Mountain region. Characteristics of this region provide insight into the range of mean annual probabilities that can reasonably represent past patterns of igneous activity. As shown in BSC (2004af), during the past approximately 11 million years, about 20 basaltic igneous events have occurred in the Crater Flat–Amargosa Valley area. These events are the basic event data used in most probability models for Yucca Mountain (CRWMS M&O, 1996aa; SAR Section 2.2.2.2.1.3). Within this area, an igneous event occurs, on average, once every million years or less (e.g., an annual recurrence of approximately 10^{-6}). However, only 1 out of these 20 past events occurred adjacent to the proposed repository site (i.e., the dike along Solitario Canyon fault) (SAR Section 2.3.11.2.1.1). This event occurrence pattern shows that there has been a roughly 1 in 1 million chance each year that a volcano occurred anywhere in the area. If a volcano did erupt in the future, this pattern indicates that there would be between a 1 in 10 (10^{-1}) to a 1 in 100 (10^{-2}) chance that this volcano would form near the proposed repository site. Therefore, a first-order estimate for mean annual probability of a future igneous event that might intersect the proposed repository is from 10^{-7} (i.e., $10^{-6} \times 10^{-1}$) to 10^{-8} (i.e., $10^{-6} \times 10^{-2}$) per year. The NRC staff's first-order confirmatory analysis, therefore, shows that the probabilistic volcanic hazard assessment and the PVHA-U mean annual probability values for intrusion into the repository by a basaltic dike of 1.7×10^{-8} and 3.1×10^{-8}, respectively, are consistent with patterns for known basaltic igneous events in the Yucca Mountain region that are less than approximately 11 million years old. The NRC staff notes that alternative probability models, including those in SAR Section 2.2.2.2.1.3 and the results of the PVHA-U, are consistent with these past patterns of basaltic igneous activity in the Yucca Mountain region.

NRC Staff's Conclusions

Uncertainties in the DOE probability estimates (i.e., probabilistic volcanic hazard assessment as provided by the PVHA and supported by the PVHA-update results) are not risk significant for the following reasons:

- The preponderance of information indicates that the mean annual probability for igneous disruption of the proposed repository by a basaltic dike (intrusive case) is on the order of 10^{-8} to 10^{-7}, and the probabilistic volcanic hazard assessment and associated update mean probabilities are within this range.

- Mean annual probability values significantly higher (i.e., 10^{-6}) or lower (i.e., 10^{-9}) than this range are not consistent with past patterns of activity in the Yucca Mountain region and therefore are not considered credible.

- The DOE (2009aa) analyses indicate that changes in dose consequences from igneous events are directly proportional to changes in igneous event probabilities, such that a change in mean annual probability from 1.7×10^{-8} to 1×10^{-7} would result in a factor of six increase in expected annual dose, as identified in SNL Appendix P (2008ag).

- Model or data uncertainties represented by an increase in mean annual probability to 10^{-7} would increase the expected annual dose from igneous events from less than tenths of millirems to no more than several millirems per year.

DOE's expert elicitation mean annual probability of 1.7×10^{-8} for the intersection of the proposed repository by a basaltic dike, and its associated uncertainty distribution for igneous event probability, has an adequate technical basis. DOE reasonably defined igneous events (igneous intrusion and volcanic eruptive) for use in the performance assessment evaluation. The events were defined adequately for DOE to calculate the probability of intrusive and extrusive (volcanic) events separately. DOE reasonably supported its probability models. DOE confirmed the results of its probability models through appropriate comparisons with the volcanic and tectonic history of the area and comparison to other published estimates of the intersection probability. The NRC staff notes that the original and explicit intent of the probabilistic volcanic hazard assessment was to develop an igneous hazard assessment for the 10,000-year time domain. However, because of the breadth of data evaluated in the PVHA expert elicitation, the probabilistic volcanic hazard assessment results are a reasonable estimate of igneous hazard at Yucca Mountain throughout the period of geologic stability (i.e., 1 million years).

The DOE estimate for the mean annual igneous intrusive event probability, 1.7×10^{-8} per year, with the uncertainty distribution described in SAR Section 2.2.2.2.1.2, is reasonable for use in its performance assessment. DOE has provided sufficient information to demonstrate that uncertainties represented by a potential increase in mean annual probability to 10^{-7} would not affect the results of the performance assessment significantly.

The NRC staff has reviewed the information in the SAR and other information DOE submitted and notes that specification of the igneous event probabilities is consistent with the guidance in the YMRP. The SAR (i) included geologic data and used that data to adequately define the igneous event and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in the igneous event probability model; (iii) provided appropriate technical bases supporting the models used and the estimated probability; and (iv) included an igneous activity analysis that was consistent with the limits on performance assessment.

2.2.1.2.2.3.2 Technical Evaluation for Seismic Event Probabilities

This section reviews and evaluates information DOE presented to estimate the probability of seismic ground motion and fault displacement at the proposed repository site. This technical evaluation of seismic event probabilities follows the review guidance provided in the YMRP Sections 2.2.1 and 2.2.1.2.2, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). As part of its technical evaluation, the NRC staff reviewed SAR Sections 2.2.2.1 and 2.3.4 and additional information provided in response to the NRC staff's request for additional information in DOE Enclosure 19 (2009ab) and DOE Enclosures 6, 7, and 8 (2009aq) and the references cited therein.

Risk Perspective

As described in TER Section 2.2.1.2.2.3.1, DOE indicated that the seismic ground motion modeling case dominates the mean annual dose for the first 10,000 years after permanent closure and that the mean annual dose from seismic ground motion is on the order of 0.001 mSv/yr [0.1 mrem/yr]. As shown in SAR Figure 2.4-18(a), the seismic ground motion modeling case constitutes most of the calculated dose after the first 2,000 years following closure. Moreover, DOE indicated that for the post-10,000-year period, the igneous intrusion modeling case and seismic ground motion modeling case provide approximately equal contributions to the total mean annual dose to the reasonably maximally exposed individual for the last 300,000 years of the time period, and that the calculated dose in each modeling case is on the order of 0.01 mSv/yr [1 mrem/yr].

Summary of DOE SAR on Seismic Event Probability

SAR Section 2.2.2.1 described DOE's overall approach to developing a seismic hazard assessment for Yucca Mountain, including fault displacement hazards. This overall approach involves the following three general steps.

1. DOE conducted an expert elicitation program in the late 1990s to develop a probabilistic seismic hazard assessment for Yucca Mountain. This assessment included a probabilistic fault displacement hazard assessment (CRWMS M&O, 1998aa; BSC, 2004bp). The probabilistic seismic hazard assessment was developed for a reference bedrock outcrop, specified as a free-field site condition with a mean shear wave velocity of 1,900 m/sec [6,233 ft/sec] and located adjacent to Yucca Mountain. This value was derived from a shear wave velocity profile of Yucca Mountain with the top 300 m [984 ft] of tuff and alluvium removed, as provided in Schneider, et al., Section 5 (1996aa).

2. DOE conditioned the probabilistic seismic hazard assessment ground motion results to constrain the large low-probability ground motions to ground motion levels that, according to DOE, are more consistent with observed geologic and seismic conditions at Yucca Mountain, as provided in BSC, ACN02 (2005aj).

3. DOE modified the conditioned probabilistic seismic hazard assessment results, using site-response modeling, to account for site-specific rock material properties of the tuff in and beneath the emplacement drifts and the site-specific rock and soil material properties of the strata beneath the geologic repository operations area (GROA). DOE used the results of the site response to develop inputs for preclosure seismic design and the preclosure seismic safety analysis as well as inputs to its postclosure TSPA calculation, as provided in BSC (2005aj) and BSC, ACN 02 (2008bl).

DOE applied these three steps equally to the preclosure seismic design and safety analyses as well as to its postclosure performance assessment. The NRC staff documented its evaluation of Step 1 in TER Section 2.5.4. The NRC staff's evaluation of those aspects of DOE's seismic hazard assessment (Steps 2 and 3) that are pertinent to postclosure performance assessment is documented in this TER chapter.

DOE Probabilistic Seismic Hazard Assessment Expert Elicitation

DOE conducted an expert elicitation on probabilistic seismic hazard assessment in the late 1990s (CRWMS M&O, 1998aa; BSC, 2004bp) on the basis of the methodology described in the Yucca Mountain Site Characterization Project (DOE, 1997aa). DOE stated that its probabilistic seismic hazard assessment methodology followed the guidance of the DOE-NRC-Electric Power Research Institute-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa). On SAR p. 2.2-67, DOE concluded that the methodology used for the probabilistic seismic hazard assessment expert elicitation is consistent with the NRC expert elicitation guidance, which is described in NUREG–1563 (NRC, 1996aa).

To conduct the probabilistic seismic hazard assessment evaluation, DOE convened two panels of experts. The first expert panel consisted of six, 3-member teams of geologists and geophysicists (seismic source teams) who developed probabilistic distributions to characterize relevant potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates incorporated random and unknown uncertainties that were specific to the regional crustal conditions of the western Basin and Range. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the probabilistic seismic hazard assessment. The two expert panels were supported by technical teams from DOE, the U.S. Geological Survey, and Risk Engineering Inc., who provided the experts with relevant data and information; facilitated the formal elicitation, including a series of workshops designed to accomplish the elicitation process; and integrated the hazard results.

The resulting ground motion hazard curves express increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves include estimates of uncertainty (see SAR Figure 2.2-9; for example, probabilistic seismic hazard assessment curves). The SAR provided probabilistic seismic hazard assessment findings on horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and peak ground velocity.

The NRC staff's review of the probabilistic seismic hazard assessment evaluation notes that DOE's expert elicitation process followed the NRC guidance provided in NUREG–1563 to quantify probabilistic seismic hazards (e.g., Cornell, 1968aa; McGuire, 1976aa). The NRC staff's review of the probabilistic seismic hazard assessment expert elicitation process is documented in TER Section 2.5.4. The basic elements of this process are (i) identification of seismic sources such as active faults or seismic zones; (ii) characterization of each of the seismic sources in terms of their activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) ground motion attenuation relationships to model the distribution of ground motions that will be experienced at the site when a given magnitude earthquake occurs at a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, including associated uncertainties. Each logic tree pathway represents one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that is representative of the seismic hazard at a site, including variability and uncertainty.

DOE provided information that shows that the probabilistic seismic hazard assessment was supported by a broad range of data, process models, empirical models, and seismological theory. Both the seismic source and ground motion characterization panels built their respective inputs to the probabilistic seismic hazard assessment on the basis of this information, which included (i) cause and effect analysis of recent instrumented events such as the 1992 M_W 5.6 Little Skull Mountain earthquake (where $\boldsymbol{M_W}$ is the moment magnitude), (ii) historic seismicity included in the probabilistic seismic hazard assessment historic catalog [as provided in CRWMS M&O Appendix G (1998aa)], (iii) ground motion parameters derived from empirical studies of worldwide ground motion data (e.g., Spudich, et al., 1999aa), and (iv) 52 exploratory trenches and excavations across fault traces with known or suspected Quaternary Period (last ~1.8 million years) fault movements (Keefer, et al., 2004aa).

Probabilistic Fault Displacement Hazard Assessment

The seismic source teams also developed a probabilistic fault displacement hazard assessment as part of the probabilistic seismic hazard assessment. To assess the postclosure performance, DOE relied on the probabilistic fault displacement hazard assessment results to support the TSPA analyses of mechanical degradation of engineered barrier systems. In SAR Section 2.3.4, DOE described how the information from the probabilistic fault displacement hazard assessment was used to develop the fault displacement abstraction and to generate inputs to the TSPA. The NRC staff's evaluation of DOE's analysis of postclosure fault displacement effects on engineered barriers is described in TER Section 2.2.1.3.2.3.2.

In the probabilistic fault displacement hazard assessment, the experts derived probabilistic fault displacement hazard curves for nine demonstration points at or near Yucca Mountain (SAR Table 2.2-15 and SAR Figure 2.2-12). These demonstration points represent a range of faulting and related fault deformation conditions in the subsurface and near the proposed surface facility sites in the geologic repository operations area, including large block bounding faults such as the Solitario Canyon Fault, smaller mapped faults within the repository footprint such as the Ghost Dance Fault, unmapped minor faults near the larger faults, fractured tuff, and intact tuff. The fault displacement hazard curves (e.g., SAR Figure 2.2-13) are analogous to seismic hazard curves, in which increasing levels of fault displacements are computed as a function of the annual probability that those displacements will be exceeded.

For the largest mapped faults at Yucca Mountain (i.e., those that form the boundary of the major fault block that comprises the Yucca Mountain geologic features), the probabilistic fault displacement hazard curves were largely based on the same detailed paleoseismic and earthquake data used to characterize these faults as potential seismic sources (CRWMS M&O, 1998aa). However, for smaller faults and fractures that were not part of the seismic source characterization, there were no established techniques available to the experts. Because of the complexity of Yucca Mountain fault analyses, the experts relied on both available information and expert judgment to develop conceptual models of distributed faulting and estimated the probabilities of secondary faulting in the repository (Youngs, et al., 2003aa; CRWMS M&O, 1998aa).

The probabilistic fault displacement hazard assessment experts derived these curves using two different methods, which DOE referred to as the displacement approach and earthquake approach. The displacement approach uses fault-specific data, such as cumulative displacement, fault length, paleoseismic data from trenches, and historic seismicity. The earthquake approach relates the frequency of the fault slip events to the frequency of earthquakes on the fault as defined in the seismic source models developed for the corresponding seismic hazard analysis.

For the displacement approach, the experts relied on direct observations of faulting, deriving the two required parameters directly from paleoseismic displacement and recurrence rate data, geologically derived slip rate data, or scaling relationships that relate displacement to fault length and cumulative fault displacement. For the earthquake approach, the experts used earthquake recurrence models from the seismic hazard analysis. For this approach, the experts assessed three probabilities:

1. The probability that an earthquake will occur. The experts derived the probability that an earthquake will occur from the frequency distribution of earthquakes for each source (fault or area) used on the seismic hazard assessment and based on geologic, historical seismic, or paleoseismic data.

2. The probability that this earthquake will produce surface rupture on the fault generating the earthquake (the primary fault where the earthquake occurs). The expert teams determined the probability of surface rupture by a statistical regression of historical earthquake and surface rupture data from the Basin and Range and focal depth calculations. In the focal depth calculations, the size and shape of the fault rupture for each earthquake (generally considered circular or elliptical) was estimated from empirical scaling relationships (e.g., Wells and Coppersmith, 1994aa). Depending on focal depth, the experts determined the surface displacement (if any) along the fault. Because the maximum surface displacement of a fault may not coincide with the demonstration point, an additional variable that randomized the rupture along the fault length was also introduced.

3. The probability that the earthquake will produce distributed surface displacement on other faults, primary or secondary. The experts determined the probability of distributed faulting by using a statistical best fit to data from Basin and Range historical ruptures in which distributed faulting was mapped after the earthquake (e.g., Pezzopane and Dawson, 1996aa) or by using slip tendency analysis (Morris, et al., 1996aa).

DOE provided information that shows the probabilistic fault displacement hazard assessment methodology used to evaluate fault displacement hazard at Yucca Mountain is reasonable. The probabilistic fault displacement hazard assessment is supported by the same broad range of data, process models, empirical models, and seismological theory used in the probabilistic seismic hazard assessment. The two methods the experts used, the displacement approach and the earthquake approach, were originally defined as the faulting-occurrence method and magnitude-occurrence method by Cornell and Toro (1992aa). The methods have been published in the scientific literature (Youngs, et al., 2003aa) and have been accepted by the NRC staff for sites other than Yucca Mountain, including the license application for the Private Fuel Storage facility in Skull Valley, Utah (Private Fuel Storage Limited Liability Company, 2001aa; NRC, 2002aa).

On the basis of the expert elicitation process performed to support the fault displacement hazard estimate in the SAR, results of the probabilistic fault displacement hazard assessment are reasonable.

Conditioning of Low Probability Ground Motions

Since completion of the probabilistic seismic hazard assessment in 1998, several studies and reports, including ones from the NRC staff (NRC, 1999aa), the Nuclear Waste Technical Review Board Panel on Natural Systems and Panel on Engineered Systems (Corradini, 2003aa), and DOE itself (e.g., BSC, 2004bj), questioned whether the very large ground motions the probabilistic seismic hazard assessment predicted at low annual exceedance probabilities (below ~10^{-6}/yr) were physically realistic. For example, strong motion recordings of acceleration and velocity that DOE scaled to the unbounded probabilistic seismic hazard assessment curve at 10^{-7} annual exceedance probability yield peak ground acceleration as high as 20 g [~640 ft/s^2] and peak ground velocities up to 1,800 cm/sec [~60 ft/s] (BSC, 2004bj). These values were based on extrapolating the expert elicitation results and are well beyond the limits of any recorded earthquake accelerations and velocities. That includes the largest recorded earthquakes worldwide. These large ground motions also are deemed physically unrealizable (e.g., Kana, et al., 1991aa) because they require a combination of earthquake stress drop, rock strain, and fault rupture propagation that cannot be sustained without wholesale fracturing of the bedrock.

In the past, probabilistic seismic hazard curves were used to estimate ground motions with annual exceedance probability to 10^{-4} or 10^{-5} (typical annual exceedance probability values for nuclear power plant design and safe shutdown earthquakes). For Yucca Mountain, however, the seismic hazard curves are extrapolated to estimate ground motions with annual exceedance probabilities as low as 10^{-8}. At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated lognormal distributions of the input ground motion attenuation models (e.g., Bommer, et al., 2004aa).

To account for these large ground motions, DOE conditioned, or reduced the hazard using two approaches. The first approach used geological observations at the repository level to develop a limiting distribution on shear strains experienced at Yucca Mountain (BSC, 2005aj). The shear-strain threshold distribution was then related to the distribution of horizontal peak ground velocity through ground motion site-response modeling. DOE used laboratory rock mechanics data, corroborated by numerical modeling, to develop the shear-strain threshold distribution. DOE derived the shear-strain levels to initiate unobserved stress-induced failure of lithophysal deposition of the Topopah Spring Tuff. DOE's site-response calculation used the random-vibration-theory-based equivalent-linear model to compute the mean motions: strains for the deaggregation earthquakes that dominate the contribution of ground motion hazard of the specified annual probability of exceedance. Later, this approach was (i) generalized to other than horizontal peak ground velocity; (ii) modified to use the inferred shear-strain threshold at the repository waste emplacement level to determine the level of ground motion not experienced at the reference rock outcrop, rather than at the waste emplacement level; (iii) refined to include variability in shear-strain levels and integration over the entire hazard curve; and (iv) updated to incorporate additional geotechnical data onsite tuff and alluvium properties in the site-response part of the approach (BSC, 2008bl).

The second approach used expert judgment (BSC, 2008bl) to develop a distribution of extreme stress drop in the Yucca Mountain vicinity, which results in strong motion far exceeding the recorded data. The distribution is based on available data (stress drop measurements and apparent stress from laboratory experiments) and interpretations. It is used in the random-vibration-theory method for point sources to develop distributions of peak ground velocity and peak ground acceleration at the reference rock outcrop. The extreme stress drop is characterized by a lognormal distribution with a median value of 400 bars and $\sigma_{\ln}$ of 0.6 (mean of 480 bars). This distribution is discretized to three values of 150, 400, and 1,100 bars with the weighting factors of 0.2, 0.6, and 0.2, respectively. This distribution is mapped into a distribution of extreme ground motion for the reference rock outcrop through the random-vibration-theory site-response modeling.

The unconditioned hazard curve, which is the annual probability of exceedance as a function of ground motion, is convolved with the distribution of extreme ground motion for the reference rock outcrop to produce the conditioned ground motion hazard of the same rock outcrop. SAR Section 1.1.5.2.5.1 stated that the conditioning is done using combined shear-strain-threshold and extreme-stress-drop approaches. However, the shear-strain-threshold conditioning has a marginal impact as compared to the extreme-stress-drop approach. For example, for an annual probability of exceedance of 10^{-8}, the shear-strain-threshold-conditioned peak ground velocity hazard is reduced from 1,200 to about 1,100 cm/sec [39.4 to about 36.1 ft/sec] or about 10 percent; the stress-drop-conditioned peak ground velocity hazard is reduced from 1,200 to about 480 cm/sec [38.4 to about 15.7 ft/sec] or about 60 percent, as identified in BSC Section A4.5.1 (2008bl). The combined conditioning has almost no impact on design basis ground motions. However, for annual probabilities of exceedance of 10^{-5}, 10^{-6}, 10^{-7}, and 10^{-8}, the impact is tremendous (SAR Section 1.1.5.2.5.1). SAR Figures 1.1-79 and 1.1-80 compare the unconditioned and conditioned peak ground acceleration and peak ground velocity mean hazard curves for the reference rock outcrop.

NRC Staff's Evaluation of the DOE SAR

The NRC staff reviewed the information in the SAR with regard to DOE's probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment and notes that these assessments are reasonable. Additionally, the staff notes the following:

- The NRC staff notes in TER Section 2.5.4, that DOE developed an expert elicitation program that is consistent with NRC guidance provided in NUREG–1563 (NRC, 1996aa). DOE's expert elicitation program was also consistent with the methodology for conducting a seismic probabilistic seismic hazard assessment elicitation as described in NUREG/CR–6372 (Budnitz, et al., 1997aa).

- The geological, geophysical, and seismological information DOE provided to the probabilistic seismic hazard assessment experts and described in the SAR and supporting documents, adequately described the site and regional seismological conditions. The information provided sufficient technical basis to support the development of expert judgment within DOE's expert elicitation program.

Thus, because DOE relied on the collective judgment of established experts, followed a procedure to elicit and document the experts' conclusions that is consistent with NRC guidance, and supported the experts' elicitation with sufficient technical and scientific information, the results of the elicitations are reasonable for use in the other portions of the SAR.

DOE reasonably defined faulting and seismicity as events without ambiguity, and used these definitions consistently in developing probability models from the probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment expert elicitations. The probabilistic estimates of faulting and seismicity were derived by DOE from appropriate geological, geophysical, and seismological data and analyses.

DOE's probabilistic seismic hazard assessment, the conditioning of the probabilistic seismic hazard assessment ground motions, and probabilistic fault displacement hazard assessment are supported by appropriate technical bases that include the expert elicitation program in which the experts considered the full range of information available when the probabilistic seismic hazard assessment was developed. During the expert elicitation process, the seismic source teams considered a range of information provided by DOE, the U.S. Geological Survey, other project-specific Yucca Mountain studies, and information published in the scientific literature. This information included data and models for the geologic and seismotectonic setting, seismic sources, historical and instrumented seismicity, earthquake recurrence, maximum magnitude, and ground motion attenuation. Detailed evaluations of this information are provided in NUREG–1762, (NRC, 2005aa). The NRC staff notes that DOE's elicitation process (see the NRC staff's evaluation in TER Section 2.5.4) for the probabilistic seismic hazard assessment was implemented in accordance with NRC guidance in NUREG–1563 (NRC, 1996aa). Because the probabilistic seismic hazard assessment expert elicitation process is consistent with NRC guidance and DOE provided supporting information, DOE's probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment programs adequately characterized the seismic and fault displacement hazards at Yucca Mountain. Additional geological, geophysical, and seismological information discovered since the elicitation was performed is consistent with the probabilistic seismic hazard assessment results (except for the overly conservative information on large ground motions at low annual exceedance, as described previously regarding the conditioning of low probability ground motions).

DOE's probabilistic fault displacement hazard assessment, the probabilistic seismic hazard assessment, the conditioning of the probabilistic seismic hazard assessment ground motions, and underlying models are adequately supported by detailed process models and empirical observations. Both the seismic source and ground motion characterization panels provided inputs to the probabilistic seismic hazard assessment. The panels considered a wide variety of geological, geophysical, and seismological information. DOE's probabilistic seismic hazard assessment report (CRWMS M&O, 1998aa) documents how the experts considered this information. Additionally, in DOE Enclosure 19 (2009ab), DOE showed that its treatment of low probability seismic ground motions in features, events, and processes screening justifications is consistent with their use in postclosure dynamic analyses and the TSPA analyses. The NRC staff thus notes (see the NRC staff's evaluation in TER Section 2.2.1.2.1) that the DOE models are consistent with other relevant features, events, and processes.

The NRC staff notes that the probabilistic seismic hazard assessment experts adequately established the probability model parameters that form the input nodes to the probabilistic fault displacement hazard assessment and probabilistic seismic hazard assessment logic tree. DOE provided an example of a partial logic tree for one of the seismic source teams in SAR Figure 2.2-21. The experts developed these probabilistic inputs to the probabilistic seismic hazard assessment by assessing the information the technical support teams provided. These inputs were based on the experts' first-hand knowledge of the information, detailed vetting of the information at the probabilistic seismic hazard assessment public workshops, and sensitivity analyses the technical support team provided as feedback to the experts. Each expert or team of experts documented its rationale for the input parameters in DOE's probabilistic seismic hazard assessment report (CRWMS M&O, 1998aa).

The NRC staff notes that the probabilistic fault displacement hazard assessment and the probabilistic seismic hazard assessment experts evaluated the uncertainties of the probability model parameters, which also form the input nodes to the probabilistic seismic hazard assessment logic tree. The experts assessed information from the technical support teams on the basis of the experts' first-hand knowledge of the information, detailed vetting of the information at the probabilistic seismic hazard assessment workshops, and sensitivity analyses the technical support team provided as feedback to the experts. Each expert or team of experts documented its rationale for the uncertainty in parameters in DOE's probabilistic seismic hazard assessment report (CRWMS M&O, 1998aa).

In addition, DOE considered new information acquired since the development of the probabilistic seismic hazard assessment in 1998. In particular, DOE's conditioned hazard curves, which reflect geological and seismological information that suggests limits on the low probability ground motions in the probabilistic seismic hazard assessment, are reasonable. DOE's conditioning approach follows basic mechanical and material behaviors consistent with the current understanding of seismological phenomena. DOE's assumption that the tectonic setting and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly in the next million years is also reasonable on the basis of the NRC staff's understanding of the seismotectonic history of the Yucca Mountain region.

Evaluation Conclusions for Seismic Event Probabilities

The NRC staff has reviewed the information in the SAR and other information submitted and concludes that the specification of the seismic event probabilities is consistent with the guidance in the YMRP. The SAR (i) included geologic data and used those data to adequately define the faulting and seismic events and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in the faulting and seismic event probability models; (iii) provided appropriate technical bases supporting the models used and the estimated probabilities; and (iv) included a seismic activity analysis that was consistent with the limits on performance assessment.

2.2.1.2.2.4 Technical Evaluation of Early Waste Package and Drip Shield Failures Event Probabilities

This section evaluates information DOE presented to estimate the probability of early failure of waste packages and the drip shield at the proposed repository site. This technical evaluation of early failure event probabilities follows the review guidance provided in YMRP Sections 2.2.1 and 2.2.1.2.2. The NRC staff reviewed SAR Sections 1.3.4, 1.5.2, 2.2.2.3, 2.3.6.6, and 2.3.6.8.4, and additional information provided in response to the NRC staff's request for additional information (DOE, 2009ac,ad) and the references cited therein.

Summary of the DOE Information

In SAR Section 2.2.2.3.1, DOE defined early failure of a waste package or drip shield as through-wall penetration of the barrier caused by the presence of a manufacturing- or handling-induced defects at a time earlier than would be predicted by mechanistic degradation models for a defect-free waste package or drip shield.

In SAR Section 2.2.2.3.2, DOE summarized the early waste package failure probability and stated that the probability values are based on the waste package fabrication and handling processes described in SAR Section 1.5.2. DOE further stated that details of technical bases for the probability estimates, including the parameters and data used and their associated uncertainties, are described in SAR Section 2.3.6.6. In SAR Section 2.2.2.3.3, DOE summarized the early drip shield failure probability and stated that the probability values are based on the drip shield fabrication and handling processes described in SAR Section 1.3.4. DOE further stated that details of technical bases for the probability estimates, including the parameters and data used and their associated uncertainties, are described in SAR Section 2.3.6.8.4.

DOE's approach for early failure probability calculations is to quantify errors in manufacturing or handling of waste packages or drip shields, respectively, and to quantify the potential that the error goes undetected prior to emplacement. In such instances, the defective waste package or drip shield is assumed to experience early failure.

DOE first systematically identified the types of errors or defects that could lead to early failure of the waste package and drip shield, respectively. It reviewed the technical literature for empirical data of similar systems and components (i.e., industrial analogues). DOE identified five industrial analogues, which can generally be described as welded metallic containers: (i) boilers and pressure vessels, (ii) nuclear fuel rods, (iii) underground storage tanks, (iv) radioactive cesium capsules, and (v) dry storage casks for spent nuclear fuel (SNF). For these analogues, DOE obtained qualitative and quantitative information on the types of manufacturing and handling errors that may occur, and their associated frequency for the occurrence, as identified in SNL Section 6.1 (2007aa).

SAR Table 2.3.6-21 identifies the specific types of defects and their occurrence rates for these analogues. From these industrial analogues, DOE developed a list of 13 generic errors or defects that could lead to early failure of welded metallic containers (SAR Section 2.3.6.6.2.1).

Given that the industrial analogues are only partly analogous to the waste package and drip shield in terms of manufacturing techniques, intended safety function, and operating environment, DOE determined that only some of the generic defects applicable to welded metallic containers are applicable to the waste package and drip shield, as identified in SAR Sections 2.3.6.6.3.1 and 2.3.6.8.4.3.1 and described in SNL Section 6.1.6 (2007aa). DOE evaluated the defect types and eliminated from further consideration those defects not applicable to the waste package and drip shield (SAR Sections 2.3.6.6.3.1 and 2.3.6.8.4.3.1). DOE considered that weld flaws, particularly in the waste package closure weld, could affect the performance of the waste package, but would not necessarily lead to early failure. It considered weld flaws as potential initiation sites for stress corrosion cracking. SAR Section 2.3.6.5 addressed weld flaws and is evaluated in TER Section 2.2.1.3.1.3.2.3.

For the waste package, DOE identified six types of defects or errors that could lead to early failure (SAR Section 2.3.6.6.3.2). These waste package errors are (i) improper base metal selection; (ii) improper weld filler material selection; (iii) improper heat treatment of the outer corrosion barrier; (iv) improper heat treatment of outer lid; (v) improper low-plasticity burnishing; and (vi) improper handling. For the drip shield, DOE identified four types of defects or errors that could lead to early failure (SAR Section 2.3.6.8.4.3.2). These drip shield errors are (i) base metal flaws; (ii) improper weld filler material; (iii) improper heat treatment; and (iv) improper handling and installation. Those defects that could lead to early failure of the waste package or failure of the drip shield were further analyzed. DOE developed event trees to identify event sequences that could lead to undetected defects or errors in the waste package and drip shield, respectively, as identified in SNL Sections 6.3 and 6.4 (2007aa). The event sequences generally consist of an equipment or process failure event, followed by human errors event(s), where the equipment or process failure is undetected or uncorrected. To quantify the probabilities for the respective event sequences, each basic event in the sequences was assigned a probability distribution. For the equipment or process failure events, the probability distribution was based on data that were generated from similar components or processes at nuclear power plants (e.g., Babcock and Wilcox, 1979aa; Blanton and Eide, 1993aa). For human reliability data, the probability distributions were taken from data for nuclear power plant activities (Swain and Guttmann, 1983aa; Benhardt, et al., 1994aa).

DOE used Monte Carlo simulations to analyze the event trees and calculate the probability distributions for event sequences that could lead to undetected errors or defects in the waste package or drip shield, respectively. DOE described end state probabilities for event sequences as lognormal distributions. DOE grouped probability distributions for all of the event sequences that could lead to the presence of an undetected defect in the waste package to calculate the overall probability that the waste package contains at least one undetected defect (i.e., the waste package early failure probability). DOE followed the same process for the drip shield to calculate the overall probability that the drip shield contains at least one undetected defect (i.e., the drip shield early failure probability). DOE clarified (2009ac) that the mean probabilities for the event sequences leading to early failure of the waste package, as reported in SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6, were incorrect and that the mean probabilities for the event sequences leading to early failure of the drip shield, as reported in SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4, were also incorrect. DOE's clarification (2009ac) provided corrected values for SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6 and SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4 and stated that the overall probability values for waste package and drip shield early failures listed in the SAR are correct. DOE described the early failure probability for the waste package as a lognormal distribution with a mean of 1.13×10^{-4} per waste package and an error factor of 8.17 (SAR Sections 2.2.2.3.2 and 2.3.6.6.3.2.7). DOE described the early failure probability for the drip shield as a lognormal distribution with a mean of 2.21×10^{-6} per drip shield and an error factor of 14 (SAR Sections 2.2.2.3.3 and 2.3.6.8.4.3.2.5).

DOE compared its probability estimates for early failure of the waste package and drip shield, respectively, with the defect-related failure rates for the industrial analogues. The failure rates for the industrial analogues for pressure vessels, nuclear fuel rods, underground storage tanks, and radioactive cesium capsules DOE cited ranged from 10^{-6} to 10^{-4} per component (SAR Table 2.3.6-21). DOE did not identify any cases of SNF casks that failed due to undetected defects after entering service.

The probability estimates for early failure of the waste package and drip shield are implemented in the TSPA model in the Early Failure Scenario Class, as described in SAR Section 2.4.1.2.3. This implementation is reviewed by the NRC staff in TER Section 2.2.1.4.1. The NRC staff's review of the implementation of the model abstraction for early failure is documented in TER Sections 2.2.1.3.1.3.1.2 and 2.2.1.3.1.3.2.4.

NRC Staff's Evaluation of DOE Information

DOE defined the early failure probability event without ambiguity. Early failure refers to through-wall penetration of the waste package or drip shield at a time earlier than the design life because of undetected manufacturing- or handling-induced defects. Early failure is distinguished from other events and processes that could lead to through-wall penetration (e.g., corrosion, tensile rupture). Further, the event definition is consistent with the barrier functions of the waste package and drip shield, respectively, as stated in SAR Table 1.9-8. Thus, the events are adequately defined.

The NRC staff reviewed DOE's assumption that the early failure probabilities for the waste package and drip shield, respectively, are equivalent to the probabilities that there are undetected manufacturing- or handling-induced defects in the respective barriers. This assumption is reasonable. Empirical observation of industrial analogues indicates that a waste package or drip shield with a manufacturing- or handling-induced defect will likely maintain some barrier capability [as identified in SNL Section 6.1 (2007aa)], and the presence of a defect, in itself, is unlikely to cause through-wall penetration of an engineered barrier. A secondary process (e.g., seismically induced loading, stress corrosion cracking) would likely need to act upon the defective barrier to cause through-wall rupture.

The NRC staff reviewed DOE's use of industrial analogues to identify the generic types of defects or errors that, if undetected, could lead to early failure. This use of industrial analogue information is reasonable. While there are no direct analogues to the waste package or drip shield, respectively, consideration of a number of generally similar analogues with different specific manufacturing techniques, intended functions, and operating environments provides an appropriate technical basis to identify the most risk-significant defects or errors that could lead to early failure. In particular, the analogues are similar to the waste package and drip shield in that they are all metallic, generally cylindrical or spherical in shape, welded, heat treated, and closed (i.e., designed to act as a barrier or container).

The NRC staff reviewed DOE's decision to eliminate from further consideration some of the generic errors and defects from the early failure probability models and notes the following:

- DOE's decision to eliminate improper weld-flux material from the early failure analyses for the waste package and drip shield is reasonable because the welding method to be employed for waste packages (SAR Section 1.9.2) and the welding method to be employed for drip shield [SNL Section 6.2.3 (2007aa)] do not use weld-flux material.

- DOE's decision to eliminate weld flaws from the drip shield and waste package early failure analyses is reasonable because (i) SAR Section 1.3.4.7 stated that the drip shield will be fully stress relieved, thus preventing stress corrosion crack initiation from weld flaws; (ii) the waste package is solution annealed to remove welding stresses, meaning that only waste package closure weld flaws will act as possible stress corrosion cracking locations; and (iii) the waste package closure weld flaws are modeled in SAR Section 2.3.6.5 as part of the stress corrosion cracking model (not part of the juvenile failure). The NRC staff confirmed that Postclosure Design Control Parameter 07-13 in SAR Table 1.9-9 stated that the drip shield will be stress relieved. The NRC staff determined that without external stress, weld flaw propagation is unlikely in a stress-relieved drip shield. Therefore, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate weld flaws from the drip shield early failure analysis.

- DOE's decision to eliminate poor weld-joint design from further early failure analyses for the waste package and drip shield is reasonable because controls specified in SAR Section 1.9.2 required extensive development and testing of waste package and drip shield joints. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-12 and 03-14 for the waste package and 07-09 and 7-10 for the drip shield) require that fabrication welds are conducted in accordance with standard nuclear industry practice, including inspection and nondestructive examination. Therefore, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate poor weld-joint design from the drip shield and waste package early failure analyses.

- DOE's decision to eliminate missing welds from further early failure analyses for the waste package and drip shield is reasonable because controls specified in SAR Section 1.9.2 require extensive inspection and nondestructive examination of the welds. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-13 and 03-14 for the waste package and 07-09 and 7-10 for the drip shield) require that fabrication welds be conducted in accordance with standard nuclear industry practice, including inspection and nondestructive examination. Literature reviews identified a low probability for missing welds and identified that the consequences of a missing weld (e.g., the waste package lid falling off) would be apparent prior to emplacement. Thus, because the Postclosure Design Control

Parameters 03-13, 03-14, 07-09 and 07-10 in SAR Table 1.9-9 require extensive inspection and nondestructive evaluation of waste packages and drip shields, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate missing welds from the drip shield and waste package early failure analyses.

- DOE's decision to eliminate mislocated welds from further early failure analyses for the waste package and drip shield is reasonable because controls specified in SAR Section 1.9.2 require extensive inspection and nondestructive examination of the welds. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-13 and 03-14 for the waste package and 07-09 and 07-10 for the drip shield) require that fabrication welds be conducted in accordance with standard nuclear industry practice, including inspection and on nondestructive examination. Thus, because the Postclosure Design Control Parameters 03-13, 03-14, 07-09 and 07-10 in SAR Table 1.9-9 require extensive inspection and nondestructive evaluation of waste packages and drip shields, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate mislocated welds from the drip shield and waste package early failure analyses.

- DOE's decision to eliminate surface contaminants (e.g., material that could enhance the corrosion rate) from further early failure analyses for the waste package and drip shield is reasonable because controls specified in SAR Section 1.9.2 require that fabrication and handling processes will limit the type and amount of surface contamination. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-21 for the waste package; 07-14 for the drip shield) require stringent controls on waste package and drip shield fabrication and handling. Therefore, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate surface contaminants from the drip shield and waste package early failure analyses.

- DOE's decision to eliminate improper low-plasticity burnishing from the drip shield early failure analysis is reasonable because the drip shield is not low-plasticity burnished. The NRC staff reviewed the description of the drip shield design in SAR Section 1.3.4.7 and confirmed that drip shield welds will not be low-plasticity burnished.

- DOE's decision to eliminate handling damage from early failure analysis for the drip shield is reasonable because the high strength-to-weight ratio of titanium makes it resilient to scratches and denting from handling-induced impacts, and because DOE will control drip shield handling. The NRC staff confirmed that Postclosure Design Control Parameter 07-14 in SAR Table 1.9-9 requires controls on drip shield handling that will minimize damage and impacts to the drip shield, and drip shield emplacement will be monitored by equipment that can detect damage. Therefore, conformance with the Postclosure Design Control Parameters provides a reasonable basis for DOE to eliminate handling damage from early failure analysis for the drip shield.

- DOE's decision to eliminate administrative or operational errors as distinct errors in the waste package and drip shield early failure analyses is reasonable because DOE implicitly incorporated such errors (e.g., failure to follow a written procedure) into the analyses of the defects that were not screened out.

The NRC staff reviewed the event trees and event sequences DOE used to calculate the probabilities for the errors that could cause early failure. The NRC staff reviewed the extent to which DOE identified key processes involved with waste package and drip shield handling and manufacturing, and whether the event sequences could estimate the undetected defect (i.e., early failure) probabilities. The NRC staff notes the following regarding DOE's event trees and event sequences used to calculate probabilities.

- The NRC staff reviewed the DOE event tree for waste package fabrication with improper base metal selection, which is shown in SNL Figure 6-9 (2007aa). In response to the NRC staff's request for additional information, DOE (2009ac) stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff notes DOE identified the key processes for this event sequence and that the event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, the NRC staff further notes that DOE's reliance on Postclosure Design Control Parameter 03-19 in SAR Table 1.9-9, which specifies the waste package outer corrosion barrier material specifications, and DOE's use of Postclosure Design Control Parameter 03-02, which requires that the waste package material be controlled by the configuration management system, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for waste package fabrication with improper weld filler material selection, which is shown in SNL Figure 6-14 (2007aa). In response to the NRC staff's request for additional information, DOE (2009ac) stated that the composition of the weld filler metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff notes that DOE identified the key processes for this event sequence. The event sequence is realistic, because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, DOE's reliance on Postclosure Design Control Parameter 03-14 in SAR Table 1.9-9, which specifies that the waste package fabrication welds shall be conducted in accordance with standard nuclear industry requirements, is reasonable; and DOE's use of Postclosure Design Control Parameter 03-02, which requires that the waste package material be controlled by the configuration management system, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for the waste package fabrication with improper heat treatment for the waste package outer shell, which is shown in SNL Figure 6-10 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SNL Section 6.3.3 (2007aa), DOE stated that the critical steps during heat treatment are moving the heated shell from the furnace to the quench tank and the subsequent quench. The event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls. DOE's reliance on Postclosure Design Control Parameter 03-16 in SAR Table 1.9-9, which specifies the waste package heat treatment conditions, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for waste package fabrication with improperly heat-treated outer lid, which is shown in SNL Figure 6-11 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SNL Section 6.3.4 (2007aa), DOE stated that the critical steps during heat treatment are moving the heated lid from the furnace to the quench tank and the subsequent quench. The event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls. DOE's reliance on Postclosure Design Control Parameter 03-16 in SAR Table 1.9-9, which specifies the waste package heat treatment conditions, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for waste package fabrication with improper low-plasticity burnishing of the closure weld, which is shown in SNL Figure 6-12 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SNL Section 6.3.5 (2007aa), DOE stated that burnishing will be performed by a dedicated, automated system, with subsequent inspection to assure that the appropriate procedures were followed. The event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls. DOE's reliance on Postclosure Design Control Parameter 03-17 in SAR Table 1.9-9, which requires process controls to ensure adequate stress relief, along with subsequent nondestructive examination, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for improper handling of the waste package, which is shown in SNL Figure 6-13 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SAR Section 2.3.6.6.3.2.5 and DOE (2009ad), DOE defined damage as any visible gouging or denting of the waste package surface that occurs between receipt and drip shield installation. Damage would be any such gouging or denting that could jeopardize the performance of the outer barrier. Because handling procedures have not been fully developed, DOE assumed that the waste package could be damaged by any one of eight generic events, each of which is analogous to fuel assembly handling events at nuclear power plants. In response to the NRC staff's request for additional information, DOE (2009ac) stated that this comparison is appropriate because fuel assemblies are handled in tightly controlled conditions similar to those expected at the repository. The event sequence is realistic because DOE reasonably described the processes and stated that handling will be accomplished under stringent controls. In this regard, DOE reliance on Postclosure Design Control Parameters 03-18, 03-21, and 03-22 in SAR Table 1.9-9 is reasonable. The Postclosure Design Control Parameters require the waste package to be handled in a controlled manner to minimize damage, including inspection for surface damage prior to emplacement and monitoring during emplacement activities. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for drip shield fabrication with out-of-specification base metal, which is shown in SNL Figure 6-16 (2007aa). In response to the NRC staff's request for additional information, DOE (2009ac) stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff notes that DOE identified the key processes for this event sequence and that the event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. DOE's reliance on Postclosure Design Control Parameter 07-09 in SAR Table 1.9-9, which specifies that the drip shield shall be fabricated in accordance with standard nuclear industry practices, including material control, is reasonable; DOE's use of Postclosure Design Control Parameter 07-01 in SAR Table 1.9-9, which requires that the drip shield materials be controlled by the configuration management system, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for drip shield fabrication with out-of-specification weld filler metal, which is shown in SNL Figure 6-18 (2007aa). In response to the NRC staff's request for additional information, DOE (2009ac) stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. DOE identified the key processes for this event sequence and that the event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, DOE's reliance on Postclosure Design Control Parameter 07-09 in SAR Table 1.9-9, which specifies that the drip shield shall be fabricated in accordance with standard nuclear industry practices, including material control and welding, is reasonable; DOE's use of Postclosure Design Control Parameter 07-01 in SAR Table 1.9-9, which requires that the drip shield materials be controlled by the configuration management system, is reasonable. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for drip shield fabrication with improper heat treatment, which is shown in SNL Figure 6-17 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SNL Section 6.4.2 (2007aa), DOE stated that the drip shield temperature during heat treatment will be monitored by calibrated thermocouples in contact with the material, and that the drip shield will be subject to a post-heat-treatment inspection to ensure that the heat-treatment procedure was properly followed. The event sequence is realistic because DOE reasonably described the processes and stated that fabrication will be accomplished under stringent controls. In this regard, DOE's reliance on Postclosure Design Control Parameters 07-09 and 07-13 in SAR Table 1.9-9 is reasonable. The Postclosure Design Control Parameters (i) require that the drip shield heat treatment be performed in a manner consistent with standard nuclear industry practice and (ii) specify the drip shield heat treatment conditions. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed the DOE event sequence for improper drip shield installation, which is shown in SNL Figure 6-19 (2007aa). The NRC staff notes that DOE identified the key processes for this event sequence. For example, in SNL Section 6.4.4 (2007aa), DOE stated that drip shields will be visually inspected at the surface facilities, that emplacement activities will be monitored by camera, and that the inspections will be independently checked and documented. In response to the NRC staff's request for additional information (RAI), DOE (2009ad) provided additional justification for the probability value of a camera not detecting improper interlocking between adjacent drip shields, and a demonstration that mechanical or equipment reliability is not a significant component of the drip shield emplacement failure analysis. The event sequence is realistic because DOE reasonably described the processes and stated that installation will be accomplished under stringent controls. DOE's reliance on Postclosure Design Control Parameters 07-02 and 07-14 in SAR Table 1.9-9 is reasonable. The Postclosure Design Control Parameters require that drip shield handling and emplacement be monitored by appropriate equipment, including an alarm, with an operator and independent inspector verifying proper installation. Therefore, the event sequence for this defect is reasonable, because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

In summary, the NRC staff has reviewed in detail the event trees DOE used to evaluate the probabilities that could lead to damage of the waste package and the drip shield. The NRC staff notes that DOE has identified the key processes involved and has implemented event sequences consistent with the guidance in the YMRP.

In SAR Sections 2.3.6.6.4.2 and 2.3.6.8.4.4.2, DOE compared its probability estimates for early failure of the waste package and drip shield, respectively, with the defect-related failure rates for the industrial analogues. For pressure vessels, nuclear fuel rods, underground storage tanks, and radioactive cesium capsules, the failure rates DOE cited are in the range of 10^{-6} to 10^{-4} per component (SAR Table 2.3.6-21), which is consistent with the calculated early failure rates for the waste package and drip shield. DOE did not identify any cases of SNF casks that failed after entering service.

The NRC staff notes that the waste package and drip shield are sufficiently similar to the industrial analogues to support a general comparison of the manufacturing- and handling-induced failure rates. In particular, the waste package, drip shield, and industrial analogues are (i) metallic, (ii) cut from sheet and formed into a cylindrical-type shape, (iii) welded, (iv) heat treated, and (v) closed/sealed (i.e., intended to act as a container or barrier).

DOE has identified key processes involved with manufacturing and handling of the respective components, and that it has developed realistic event sequences to calculate the early failure probabilities. DOE's model support for estimating waste package and drip shield early failure probabilities is reasonable. Predicted early failure rates for the waste package and drip shield are close to those of the industrial analogues, as discussed previously.

DOE developed event sequences to calculate the probabilities for undetected errors or defects (i.e., the early failure probabilities) in the waste package and drip shield, respectively. The event sequences generally consist of an equipment or process failure (e.g., probability that a motorized valve fails to open on demand), followed by human error(s), where the equipment or process failure is undetected (e.g., probability that the responsible technician does not detect the failure of the valve to open) or uncorrected. As described in SAR Sections 2.3.6.8.4.2 and

2.3.6.6.2, each event in the event sequences was assigned a probability distribution that was obtained from external data sources, as identified in SNL Section 4.1 (2007aa).

The external data used to establish the probability distributions for key processes and events in waste package and drip shield manufacturing and handling come from nuclear power plant activities. For the equipment or process failure events, DOE used reliability data that were generated from similar components or activities at nuclear power plants (e.g., Babcock and Wilcox, 1979aa; Blanton and Eide, 1993aa). For human error events, the probability distributions for these events were taken from nuclear power plant human reliability analyses (Swain and Guttmann, 1983aa; Benhardt, et al., 1994aa).

The NRC staff reviewed the external data to determine whether such data could be used to quantify the reliability of events and processes associated with manufacturing and handling of the waste package and drip shield. The external data come from reliable, reputable sources that are widely accepted in the nuclear industry, as identified in SNL Section 4.1 (2007aa). Further, NRC staff noted that SAR Section 1.9.2 specified rigorous controls on the manufacturing and handling of the waste package and drip shield, including conformance with nuclear industry standards and codes (e.g., American Society of Mechanical Engineers Boiler and Pressure Vessel Code). The NRC staff also recognized that the Design Control Parameters in the SAR are subject to management systems. On the basis of the nature of the data DOE used, and DOE's adherence to industry codes and standards, DOE has appropriately used data from nuclear power plants to establish the probability distributions for key processes and events in waste package and drip shield manufacturing and handling. Therefore, probability model parameters have been adequately established, consistent with the guidance in the YMRP.

DOE represented each basic event in an event sequence that can lead to an undetected defect (i.e., early failure) by a lognormal distribution. For the human error events, the external human reliability data DOE cited specify lognormal distributions with particular mean values and error factors (presented in SAR Table 2.3.6-22). For equipment or process failure events, the reliability data DOE cited typically specify only point (mean) values. As a result, DOE assigned an error factor to the probability data given in the literature as point (mean) values, as identified in SNL Section 5.3 (2007aa). DOE assumed that this point (mean) value is the mean of an unspecified probability distribution and that it is therefore appropriate to characterize the reliability with any reasonable, probability distribution. DOE used the lognormal distribution to be consistent with the human reliability data.

DOE used Monte Carlo simulations to calculate the probability distributions for the end states of the event sequences that could lead to early failure. Because the probability distributions for the basic events in the sequences may have different error factors, DOE stated that the mean value of the probability distribution for the end state of the sequence is not just a simple product of the mean of each basic event in the sequence, as identified in SNL Section 6.5.1 (2007aa). As described in DOE (2009ac), the probability distributions for all of the event sequences that could lead to an undetected defect in the waste package were combined to give the overall probability that the waste package has at least one undetected defect, which is assumed to be equivalent to the waste package early failure probability. The same was done for the drip shield. DOE ran 90,000 realizations to obtain the probability distributions for early failure of the waste package and drip shield, respectively, as identified in SNL Section 6.5.1 (2007aa).

The NRC staff reviewed the treatment of uncertainty in the early failure probability calculations. DOE has established reasonable uncertainty distributions for the events in the event sequences that can lead to undetected defects (i.e., early failure). The lognormal distributions used for human reliability events are consistent with common practice (Swain and Guttmann, 1983aa). For those events given in the literature as a mean failure rate, DOE has assumed an uncertainty range that is consistent with human reliability events and does not overestimate the reliability of components and processes. Further, DOE has reasonably propagated uncertainty through the early failure probability calculations for the waste package and drip shield, respectively. Use of Monte Carlo simulation is appropriate to ensure that the output is unbiased. DOE ran a sufficient number of realizations with Monte Carlo sampling to support its probability estimates. In summary, the NRC staff notes that uncertainty in event probability has been addressed because DOE used reasonable uncertainty distributions, the assumptions that DOE used do not overestimate the reliability of components and processes, and DOE adequately propagated uncertainty.

The NRC staff notes that the probability distributions and values DOE provided for the probabilities of waste package and drip shield early failure are lognormal distributions. There is a mean of 1.13×10^{-4} failures per waste package and an error factor of 8.17 (SAR Section 2.3.6.6.3.2.7). There is a mean of 2.21×10^{-6} failures per drip shield and an error factor of 14 (SAR Section 2.3.6.8.4.3.2.5). Use of these distributions and values in DOE's Yucca Mountain TSPA is reasonable.

Evaluation Conclusions for Early Waste Package and Drip Shield Event Probabilities

The NRC staff has reviewed the information in the SAR and other information submitted and notes that the specification of the early waste package and drip shield event probabilities are consistent with the guidance in the YMRP. The SAR (i) included information on the design of the engineered barrier system to adequately define the waste package and drip shield early failure events and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in waste package and drip shield early failure analyses; and (iii) provided appropriate technical bases supporting the analyses used and the estimated probabilities.

2.2.1.2.2.5 NRC Staff Conclusions

NRC staff notes that the DOE description of the identification of events with probabilities greater than 10^{-8} per year is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the performance assessment.

2.2.1.2.2.6 References

Babcock and Wilcox. 1979aa. "Records Investigation Report Related to Off-Chemistry Welds in Material Surveillance Specimens and Response to IE Bulletin 78-12 and 78-12A—Supplement." Mt. Vernon, Indiana: Babcock and Wilcox.

Benhardt, H.C., S.A. Eide, J.E. Held, L.M. Olsen, and R.E. Vail. 1994aa. "Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities." WSRC–TR–93–581. Aiken, South Carolina: Savannah River Site, Westinghouse Savannah River Company.

Blanton, C.H. and S.A. Eide. 1993aa. "Savannah River Site Generic Data Base Development (U)." WSRC–TR–93–262. Aiken, South Carolina: Westinghouse Savannah River Company.

Bommer, J.J., N.A. Abrahamson, F.O. Strasser, A. Pecker, P.-Y. Bard, H. Bungum, F. Cotton, D. Fäh, F. Sabetta, F. Scherbaum, and J. Struder. 2004aa. "The Challenge of Defining Upper Bounds on Earthquake Ground Motions." *Seismological Research Letters*. Vol. 75. pp. 82–95.

Boyle, W.J. 2008aa. "Transmittal of Report: Probabilistic Volcanic Hazard Analysis Update (PVHA-U) for Yucca Mountain, Nevada." Letter (October 17) to Director, DHLWRS, NRC. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

BSC. 2008bl. "Supplemental Earthquake Ground Motion Input for a Geologic Repository at Yucca Mountain, NV." MDL–MGR–GS–000007. Rev. 00. ACN 01, ACN 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005aj. "Peak Ground Velocities for Seismic Events at Yucca Mountain, Nevada." ANL–MGR–GS–000004. Rev. 00. ACN 01, ACN 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004af. "Characterize Framework for Igneous Activity at Yucca Mountain, Nevada." ANL–MGR–GS–000001. Rev. 02. ACN 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bi. "Yucca Mountain Site Description." TDR–CRW–GS–000001. Rev. 02 ICN 01. ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bj. "Technical Basis Document No. 14: Low Probability Seismic Events." Rev. 1. MOL 20000510.0175. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bp. "Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada." ANL–CRW–GS–000003. Rev. 00. MOL20000510.0175. DOC20040223.0007. Las Vegas Nevada: Bechtel SAIC Company, LLC.

Budnitz, R.J., G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris. 1997aa. NUREG/CR–6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts—Main Report." Vol. 1. Washington, DC: NRC.

CNWRA and NRC. 2008aa. "Risk Insights Derived From Analyses of Model Updates in the Total-system Performance Assessment Version 5.1 Code." ML082240343. San Antonio, Texas: CNWRA.

Corradini, M.L. 2003aa. "Board Comments on February 24, 2003 Panel Meeting on Seismic Issues." Letter (June 27) to Dr. Margaret S.Y. Chu, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: United States Nuclear Waste Technical Review Board.

Cornell, C.A. 1968aa. "Engineering Seismic Risk Analysis." *Bulletin of the Seismological Society of America.* Vol. 58. pp. 1,583–1,606.

Cornell, C.A. and G. Toro. 1992aa. *Techniques for Determining Probabilities for Geologic Events and Processes.* R.L. Hunter and C.J. Mann, eds. New York City, New York: Oxford University Press.

Costa, F. 2008aa. “Residence Times of Silicic Magmas Associated With Calderas.” Developments in Volcanology, Vol. 10. Caldera Volcanism: Analysis, Modeling, and Response. J. Gottsmann and J. Marti, eds. Amsterdam, The Netherlands: Elsevier. pp. 1–55.

CRWMS M&O. 1998aa. “Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada.” WBS 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O. 1996aa. “Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada.” BA0000000–01717–2200–00082. Rev. 0. Las Vegas, Nevada: CRWMS M&O.

Detournay, E., L.G. Mastin, J.R.A. Pearson, A.M. Rubin, and F.J. Spera. 2003aa. “Final Report of the Igneous Consequences Peer Review Panel.” DN2000219072. MOL20031014:0097. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

DOE. 2009aa. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.2.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.2, Set 2.” Letter (January 27) J.R. Williams to J.H. Sulima (NRC). ML090280281. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009ab. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2.” Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550099. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009ac. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.6.6, 2.3.6.8.4, and 2.2.2.3), Safety Evaluation Report Volume 3, Chapter 2.2.1.2.2, Set 2.” Letter (January 9) J.R. Williams to J.H. Sulima (NRC). ML090120301. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009ad. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.6.6, 2.3.6.8.4, and 2.2.2.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.2, Set 2.” Letter (January 16) J.R. Williams to J.H. Sulima (NRC). ML090210465. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009aq. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 1.1.10, 1.2.2, 1.1.5.2, and 1.1.5.3), Safety Evaluation Report Vol. 2, Chapter 2.1.1.1, Set 1.” Letter (January 12) J.R. Williams to C. Jacobs (NRC). Enclosures (10). ML090270750. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009av. DOE/RW–0573, “Safety Analysis Report Yucca Mountain Repository License Application.” Rev. 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bd. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.2.2.), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.2, Set 3." Letter (July 20) J.R. Williams to J.H. Sulima (NRC). ML092010472. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008ab. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 0. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 1997aa. "Topical Report YMP/TR–002–NP: Methodology To Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain." Rev. 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Fleck, R.J., B.D. Turrin, D.A. Sawyer, R.G. Warren, D.E. Champion, M.R. Hudson, and S.A. Minor. 1996aa. "Age and Character of Basaltic Rocks of the Yucca Mountain Region, Southern Nevada." Journal of Geophysical Research. Vol. 101, No. B4. pp. 8,205–8,227.

Hill, B.E. and C.B. Connor. 2000aa. "Technical Basis for Resolution of the Igneous Activity Key Technical Issue." San Antonio, Texas: CNWRA.

Hill, B.E. and J.A. Stamatakos. 2002aa. "Evaluation of Geophysical Information Used To Detect and Characterize Buried Volcanic Features in the Yucca Mountain Region." San Antonio, Texas: CNWRA.

Ho, C.H. and E.I. Smith. 1998aa. "A Spatial-Temporal/3-D Model for Volcanic Hazard Assessment: Application to the Yucca Mountain Region, Nevada." *Mathematical Geology*. Vol. 30, No. 5. pp. 497–510.

Ho, C.H. and E.I. Smith. 1997aa. "Volcanic Hazard Assessment Incorporating Expert Knowledge: Application to the Yucca Mountain Region, Nevada, U.S.A." *Mathematical Geology*. Vol. 29. pp. 615–627.

Ho, C.H., E.I. Smith, and D.L. Keenan. 2006aa. "Hazard Area and Probability of Volcanic Disruption of the Proposed High-Level Radioactive Waste Repository at Yucca Mountain, Nevada, USA." *Bulletin of Volcanology*. Vol. 69. pp. 117–123.

Kana, D.D., B.H.G. Brady, B.W. Vanznat, and P.K. Nair. 1991aa. NUREG/CR–5440, "Critical Assessment of Seismic and Geotechnical Literature Related to a High-Level Nuclear Waste Underground Repository." Washington DC: NRC.

Keefer, W.R., J.W. Whitney, and E.M. Taylor. 2004aa. "Quaternary Paleoseismology and Stratigraphy of the Yucca Mountain Area, Nevada." U.S. Geological Survey Professional Paper 1689. Denver, Colorado: U.S. Geological Survey.

Magsino, S.L., C.B. Connor, B.E. Hill, J.A. Stamatakos, P.C. LaFemina, D.A. Sims, and R.H. Martin. 1998aa. "CNWRA Ground Magnetic Surveys in the Yucca Mountain Region, Nevada (1996–1997)." CNWRA 98-001. San Antonio, Texas: CNWRA.

McGuire, R.K. 1976aa. "FORTRAN Computer Program for Seismic Risk Analysis." U.S. Geological Survey Open-File Report 76-67. Reston, Virginia: U.S. Geological Survey.

Morris, A.P., D.A. Ferrill, and D.B. Henderson. 1996aa. "Slip-Tendency Analysis and Fault Reactivation." *Geology.* Vol. 24. pp. 275–278.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC. 2002aa. "Consolidated Safety Evaluation Report Concerning the Private Fuel Storage Facility." ML020850233. Docket No. 72-22. Washington, DC: NRC.

NRC. 1999aa. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Rev. 2. Washington, DC: NRC.

NRC. 1996aa. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC.

O'Leary, D.W., E.A. Mankinen, R.J. Blakely, V.E. Langenheim, and D.A. Ponce. 2002aa. "Aeromagnetic Expression of Buried Basaltic Volcanoes Near Yucca Mountain, Nevada." U.S. Geological Survey Open-File Report 02-020. Denver, Colorado: U.S. Geological Survey.

Perry, F.V., A.H. Cogbill, and R.E. Kelley. 2005aa. "Uncovering Buried Volcanoes at Yucca Mountain." *Eos, Transactions, American Geophysical Union*. Vol. 86, No. 47. pp. 485–488.

Pezzopane, S.K. and T.E. Dawson. 1996aa. "Fault Displacement Hazard: A Summary of Issues and Information in Seismotectonic Framework and Characterization of Faulting at Yucca Mountain, Nevada." J.W. Whitney, report coord. U.S. Geological Survey Milestone Report 3GSH100M, Chapter 9. MOL.19970129.0041. Denver, Colorado: U.S. Geological Survey.

Private Fuel Storage Limited Liability Company. 2001aa. "Safety Analysis Report for the Private Fuel Storage Facility." Rev. 22. Docket No. 72-22. LaCrosse, Wisconsin: Private Fuel Storage Limited Liability Company.

Sawyer, D.A., R.J. Fleck, M.A. Lanphere, R.G. Warren, D.E. Broxton, and M.R. Hudson. 1994aa. "Episodic Caldera Volcanism in the Miocene Southwestern Nevada Volcanic Field: Revised Stratigraphic Framework, 40Ar/39Ar Geochronology, and Implications for Magmatism and Extension." Geological Society of America Bulletin. Vol. 106, No. 10. pp. 1,304–1,318.

Schneider, J.F., N.A. Abrahamson, and T.C. Hanks. 1996aa. "Ground Motion Modeling of Scenario Earthquakes at Yucca Mountain: Final Report for Activity 8.3.1.17.3.3." MOL.19980617.0477. Las Vegas, Nevada: Yucca Mountain Project.

Smith, E.I., D.L. Keenan, and T. Plank. 2002aa. "Episodic Volcanism and Hot Mantle: Implications for Volcanic Hazard Studies at the Proposed Nuclear Waste Repository at Yucca Mountain, Nevada." *GSA Today*. Vol. 12, No. 4. pp. 4–11.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ah. "Probabilistic Volcanic Hazard Analysis Update (PVHA-U) for Yucca Mountain, Nevada." Rev. 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aa. "Analysis of Mechanisms for Early Waste Package/Drip Shield Failure." ANL–EBS–MD–000076. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ae. "Characterize Eruptive Processes at Yucca Mountain, Nevada." ANL–MGR–GS–000002. Rev. 03. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

Spudich, P., W.B. Joyner, A.G. Lindh, D.M. Boore, B.M. Margaris, and J.B. Fletcher. 1999aa. "SEA99: A Revised Ground Motion Prediction Relation for Use in Extensional Tectonic Regimes." *Bulletin of the Seismological Society of America*. Vol. 89, No. 5. pp. 1,156–1,170.

Stamatakos, J.A., S. Biswas, and M. Silver. 2007aa. "Supplemental Evaluation of Geophysical Information Used To Detect and Characterize Buried Volcanic Features in the Yucca Mountain Region." San Antonio, Texas: CNWRA.

Stamatakos, J.A., C.B. Connor, and R.H. Martin. 1997ab. "Quaternary Basin Evolution and Basaltic Volcanism of Crater Flat, Nevada, From Detailed Ground Magnetic Surveys of the Little Cones." *Journal of Geology*. Vol. 105. pp. 318–330.

Swain, A.D. and H.E. Guttmann. 1983aa. NUREG/CR–1278, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications Final Report." Washington, DC: NRC.

Wells, D.L. and K.J. Coppersmith. 1994aa. "New Empirical Relationships Among Magnitude, Rupture Length, Rupture Width, Rupture Area, and Surface Displacement." *Bulletin of the Seismological Society of America.* Vol. 84. pp. 974–1,002.

Youngs, R.R., W.J. Arabasz, R.E. Anderson, A.R. Ramelli, J.P. Ake, D.B. Slemmons, J.P. McCalpin, D.I. Doser, C.J. Fridrich, F.H. Swan, III, A.M. Rogers, J.C. Yount, L.W. Anderson, K.D. Smith, R.L. Bruhn, P.L. Knuepfer, R.B. Smith, C.M. dePolo, D.W. O'Leary, K.J. Coppersmith, S.K. Pezzopane, D.P. Schwartz, J.W. Whitney, S.S. Olig, and G.R. Toro. 2003aa. "A Methodology for Probabilistic Fault Displacement Hazard Analyses (PFDHA)." *Earthquake Spectra.* Vol. 19, No. 1. pp. 191–219.

(Intentionally Left Blank)

CHAPTER 4

2.2.1.3.1 Degradation of Engineered Barriers

2.2.1.3.1.1 Introduction

This chapter addresses the chemical degradation of the drip shield and waste packages stored in the repository drifts. The drip shield and the waste packages are engineered barriers, a subset of the Engineered Barrier System. The general functions of the Engineered Barrier System are to (i) prevent or significantly reduce the amount of water that contacts the waste, (ii) prevent or significantly reduce the rate at which radionuclides are released from the waste, and (iii) prevent or significantly reduce the rate at which radionuclides are released from the engineered barrier system to the Lower Natural Barrier [Safety Analysis Report (SAR) Section 2.1.1.2 (DOE, 2008ab)]. The engineered barrier system consists of the emplacement drift, the drip shield, the waste package, the naval spent nuclear fuel structure, the waste form and waste package internals (e.g., transportation, aging, and disposal canisters), the waste package pallet, and invert features (SAR Figure 2.1-7).

In the postclosure performance assessment, the U.S. Department of Energy (DOE) evaluated whether the ability of the engineered barrier system components to perform their barrier functions could be compromised by features, events, and processes (FEPs) that degrade their physical structure. In particular, DOE considered that the engineered barrier system components were subject to mechanical degradation caused by seismic ground motion (SAR Section 2.3.4). The U.S. Nuclear Regulatory Commission (NRC) staff's review of DOE's Total System Performance Assessment (TSPA) models for mechanical degradation of the engineered barrier system is found in TER Section 2.2.1.3.2. The other class of engineered barrier system degradation that DOE considered in the postclosure performance assessment was chemical degradation, or corrosion, caused by reactions between the engineered barrier system materials and the environment. In SAR Section 2.3.6, DOE described the TSPA model abstractions for chemical degradation of the drip shield and the waste package outer barrier. This chapter reviews DOE's TSPA model abstractions for chemical degradation of the drip shield and the waste package outer barrier.

2.2.1.3.1.2 Evaluation Criteria

The NRC staff's review of model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for degradation of engineered barriers, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, the dose calculation should

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure

- Include in the performance assessment those features events and processes used in the performance assessment for the initial 10,000-year period

For this model abstraction of degradation of engineered barriers, 10 CFR 63.342(c)(3) further calls for the assessment of the effects of general corrosion on engineered barriers and specifies either the use of a constant representative rate throughout the period of geologic stability or a distribution of rates correlated to other parameters. DOE elected to use a distribution of corrosion rates in its SAR; thus this method is reviewed for the post-10,000-year period.

The NRC staff's review of the SAR and supporting information follows the guidance in Yucca Mountain Review Plan (YMRP) Sections 2.2.1.3.1, Degradation of Engineered Barriers, (NRC, 2003aa) as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions provide guidance on information the NRC staff could use to evaluate the performance assessment. Following the guidance, the NRC staff's review of DOE's abstraction of degradation of engineered barriers considered five criteria:

1. System description and model integration are adequate.
2. Data are sufficient for model justification.
3. Data uncertainty is characterized and propagated through the abstraction.
4. Model uncertainty is characterized and propagated through the abstraction.
5. Model abstraction output is supported by objective comparisons.

The NRC staff review used a risk-informed approach and the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of degradation of engineered barriers important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE, and on NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.1.3 Technical Evaluation

DOE's models for chemical degradation of the engineered barrier systems focus on the drip shield and the waste package outer barrier, respectively. Consistent with the YMRP guidance, the NRC staff performed a risk-informed, performance-based review, focusing on those aspects of the DOE models for chemical degradation of the drip shield and the waste packages that are most important to the calculations of barrier capability. DOE concluded that seepage flux is the primary source of water that may react with the engineered barrier system components (SAR Section 2.3.7.12.1). In the DOE model for flow of seepage water through the engineered barrier system, the water must first pass through the drip shield and then through the waste package before contacting and mobilizing the waste form. As such, this chapter first concentrates on DOE's models for chemical degradation of the drip shield and then addresses DOE's models for chemical degradation of the waste package.

2.2.1.3.1.3.1 Drip Shield Degradation

The drip shield, which DOE described in SAR Section 1.3.4.7, is an engineered metal barrier designed to divert water that enters the drift and prevent it from contacting the waste package. DOE stated that the drip shield will be fabricated from Titanium Grade 7 (UNS R52400). Titanium Grade 7 is a commercially pure titanium alloy with the addition of a small amount of palladium (approximately 0.2 wt%) to enhance its corrosion resistance. The drip shield structural supports will be fabricated from Titanium Grade 29 (UNS R56404), which is a titanium alloy composed of approximately 6 wt% aluminum and 4 wt% vanadium for strength, plus approximately 0.1 wt% ruthenium for corrosion resistance.

In developing the postclosure performance assessment analysis, DOE evaluated a number of features, events, and processes (in SAR Table 2.2-5) related to chemical degradation of the drip shield, including

- General corrosion of the drip shields [features, events, and processes (FEP) 2.1.03.01.0B]
- Stress corrosion cracking of the drip shields (FEP 2.1.03.02.0B)
- Localized corrosion of the drip shields (FEP 2.1.03.03.0B)
- Hydride cracking of the drip shields (FEP 2.1.03.04.0B)
- Microbially influenced corrosion (MIC) of the drip shields (FEP 2.1.03.05.0B)
- Early failure of the drip shields (FEP 2.1.03.08.0B)
- Oxygen embrittlement of the drip shields (FEP 2.1.06.06.0B)
- Creep of metallic materials in the drip shield (FEP 2.1.07.05.0B)

- Localized corrosion on drip shield surfaces due to deliquescence (FEP 2.1.09.28.0B)

- Thermal sensitization of the drip shields (FEP 2.1.11.06.0B)

With the exception of general corrosion and early failure of drip shields, these features, events, and processes were screened out from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these features, events, and processes from the performance assessment is addressed in TER Section 2.2.1.2.1.

With respect to the features, events, and processes that are included in the performance assessment, DOE described general corrosion of the drip shield as the uniform thinning of both the Titanium Grade 7 drip shield plates and the Titanium Grade 29 structural supports (SAR Section 2.3.6.8.1.1). In SAR Section 2.2.2.3, DOE defined drip shield early failure as through-wall penetration caused by manufacturing- and handling-induced defects, at a time earlier than would be expected for a nondefective drip shield.

In the TSPA analysis, DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure, as detailed in DOE Enclosure 1 (2009dg). The Total System Performance Assessment (TSPA) analysis calculates that few drip shields will fail within 12,000 years after repository closure. Therefore, the probability of waste package breach by localized corrosion is low in the DOE model. Following 12,000 years after repository closure, DOE calculated that there is a low probability for conditions in the drift to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package.

Other than for localized corrosion, the integrity of the drip shield does not have a significant effect on the DOE model abstractions for chemical degradation of the waste package. In the TSPA Nominal Modeling Case, DOE's models for general corrosion and stress corrosion cracking of the waste package conservatively assume aqueous degradation conditions, even for the intact drip shield. In the Seismic Ground Motion Modeling Case in the TSPA analysis, the presence of the drip shield does have some effect on stress corrosion cracking of the waste package because DOE calculated that the waste package under an intact drip shield will have a greater likelihood of being damaged under low-probability seismic ground motion events than it would under the assumption of a failed drip shield condition. This is because an intact drip shield permits unobstructed free movement of the waste package, thereby potentially causing damage as waste packages strike one another (SAR Section 2.3.4.5). Under a collapsed drip shield event, waste packages are constrained from significant movement and unable to strike or bump into each other. Consequently, a waste package under an intact drip shield is more susceptible to stress corrosion cracking if a low probability seismic event occurs that imparts the required energy for the waste packages to strike each other. Nevertheless, DOE calculated that the probability of a seismic ground motion with sufficient magnitude to damage the waste package is so low, even in the Seismic Ground Motion Modeling Case, that the presence of the drip shield has an insignificant effect on the postclosure performance assessment beyond 12,000 years after repository closure, as described in DOE Enclosure 5 (2009cn).

The NRC staff's reviews of the DOE model abstractions for general corrosion and early failure of the drip shield are presented in the following sections. Because the presence of the drip shield is important for the DOE calculations that localized corrosion of the waste package is unlikely within 12,000 years after repository closure, the NRC staff focused on those aspects of the models that were most important to the DOE calculations of the drip shield lifetime.

2.2.1.3.1.3.1.1 Drip Shield General Corrosion

In SAR Section 2.3.6.8.1, DOE described the model for general corrosion of the drip shield that was implemented in the TSPA. The drip shield is constructed of titanium alloys that are assumed to be highly corrosion resistant because of their passivity. Passivity refers to a state in which metals and alloys lose their chemical reactivity under certain environmental conditions. The passive state is generally attributed to the presence of a thin, protective oxide film on the metal surface. Because the maintenance of the passive state is important to the corrosion performance of the drip shield, the NRC staff first reviewed the drip shield's long-term passive film stability in the repository conditions. The NRC staff then reviewed the model abstraction used to calculate the drip shield general corrosion rate in the TSPA model.

Drip Shield's Long-Term Passive Film Stability

In BSC Section 1.1 (2004as), DOE presented literature references (Pourbaix, 1974aa; Schutz and Thomas, 1987aa) which indicated that the passive films on titanium alloys are stable over wide ranges of chemical potential and pH, and that, should the passive film rupture, titanium has a strong tendency for repassivation in the type of oxidizing conditions that are expected in the repository. DOE, however, also cited literature references (e.g., Lorenzo de Mele and Cortizo, 2000aa; Brossia, et al., 2001aa; Brossia and Cragnolino, 2000aa, 2001ab, 2004aa; Pulvirenti, et al., 2002aa, 2003aa) which indicated that dissolved fluoride in brine solutions can increase the general corrosion rates for titanium alloys and possibly compromise the stability of the passive film . Therefore, DOE evaluated the uncertainty in long-term drip shield passive film persistence associated with possible passive film degradation by fluoride-bearing seepage water brines, as described in BSC Section 6.5.7 (2004as).

In BSC Section 6.5.7.2 (2004as), DOE reviewed and analyzed passive film instability. DOE cited literature references that described the onset of localized corrosion on titanium specimens that were exposed to fluoride shortly after the passive film was manually removed by polishing (e.g., Brossia and Cragnolino, 2000aa, 2001ab; Brossia, et al., 2001aa). When the specimens were in an oxidizing environment, for as little as 4 days prior to fluoride exposure the specimens exhibited resistance to fluoride attack (Lorenzo de Mele and Cortizo, 2000aa). DOE stated that it expects the drip shield to have an extended period of dry thermal oxidation between the time the repository is closed and the time at which seepage water may fall onto the drip shield, as described in BSC Section 6.5.7 (2004as). Even for thermally oxidized Titanium Grade 7 specimens, however, passive film instability in fluoride-rich solution with low pH (~4) has been observed (Lian, et al., 2005aa). However, DOE concluded that such conditions are not representative of the environment expected in the repository, as outlined on SNL p. 6-408 (2008ac). DOE stated that even if seepage water brines in the repository contain fluoride, high concentrations of other species will also be present that will suppress or neutralize any fluoride attack. DOE identified studies of alloys with similar composition to Titanium Grade 7 in environments with temperatures up to 177 °C [351 °F], pH as low as 1, and fluoride, along with other species such as calcium, magnesium chloride, and silicate (Thomas and Bomberger, 1983aa; Schutz and Grauman, 1986aa). The studies showed that the titanium had a high passive film persistence, which was attributed to calcium reducing the fluoride ion solubility by precipitation of calcium fluoride, as well as the displacement of fluoride from absorption on the passive film by other species. Moreover, DOE presented its own test results in BSC Section 6.5 (2004as) in which Titanium Grade 7 specimens showed no evidence of passive film instability after 5 years' exposure to simulated concentrated water, which contained fluoride, as well as chloride, silica, sulfate, nitrate, and bicarbonate (composition given in SAR Table 2.3.6-1). Therefore, DOE concluded that the drip shield passive film will be stable during the postclosure period given the expected composition of seepage water brines, as described on SNL p. 6-410 (2008ac).

Summary of NRC Staff's Review for Drip Shield's Long-Term Passive Film Stability

The NRC staff reviewed DOE's assessment of drip shield passivity. On the basis of the information DOE provided in BSC Section 6.5.7 (2004as), the NRC staff notes that uncertainty in the long-term persistence of the titanium passive film is primarily related to potential passive film degradation by fluoride-bearing brines. There was no evidence of localized corrosion of Titanium Grade 7 exposed to fluoride-bearing simulated concentrated water for 5 years and thus that the passive film was stable when in contact with a brine having this composition. The NRC staff notes that the extent of titanium passive film degradation will generally decrease with decreasing fluoride concentration in the brines (Brossia, et al., 2001aa). Analyses by the NRC staff (Pabalan, 2010aa) indicate that the fluoride concentration in simulated concentrated water is greater than would be expected in water that would contact the drip shield in repository conditions because other species in the waters, such as calcium, can precipitate fluoride ions out of solution, thus limiting the free-chloride concentration. Therefore, the NRC staff notes that the 5-year testing of the Titanium Grade 7 specimens in fluoride-bearing simulated concentrated water supports the DOE assumption of passive film stability in repository conditions. The literature references DOE cited (Thomas and Bomberger, 1983aa; Schutz and Grauman, 1986aa) further show that other species in the waters, such as calcium, can protect the passive film by causing fluoride ions to precipitate out of solution. Finally, independent analyses the NRC staff conducted (Lin, et al., 2003aa; Codell and Leslie, 2006aa) also suggest that seepage water brines that may contact the drip shield will have insufficient fluoride concentration to significantly affect passive film stability on the drip shield titanium alloys. On the basis of this information, the NRC staff notes DOE's assumption that the drip shield is protected by a passive oxide film during the postclosure period is reasonable.

Drip Shield General Corrosion Conceptual Model

In SAR Section 2.3.6.8.1, DOE described the conceptual model for general corrosion of the drip shield that was implemented in the TSPA. In the DOE model, corrosion begins at the time of repository closure and progresses at a constant rate over time. DOE assumed aqueous conditions in the drift and also that the general corrosion rate is independent of in-drift environmental conditions (e.g., temperature, relative humidity).

The NRC staff requested DOE's technical basis for assuming that the general corrosion rate of the drip shield is independent of temperature. In DOE Enclosure 3 (2009cn), DOE stated that at the start of the general corrosion process the corrosion rates of titanium alloys are temperature dependent. However, over time, the corrosion rates at different temperatures tend to converge. DOE observed a noticeable trend of increasing corrosion rate with increasing temperature for Titanium Grade 7 specimens tested in the range of 50 to 110 °C [122 to 230 °F] after 4 weeks' exposure, but DOE also observed that the corrosion rate was less temperature dependent after 8 weeks (Hua and Gordon, 2004aa). Further, DOE referenced 3-year corrosion tests of titanium plus 0.2 weight percent palladium, which has nearly the same composition as Titanium Grade 7 in the temperature range of 90 to 200 °C [194 to 392 °F] in a pH 4.9 chloride-sulfate brine (Smailos and Köster, 1987aa). DOE concluded that the corrosion rates initially showed some temperature dependence, but were effectively identical within 3 years as shown in Smailos and Köster Figure 1 (1987aa).

Summary of NRC Staff's Review for Drip Shield General Corrosion Conceptual Model

The NRC staff reviewed DOE's conceptual model for general corrosion of the drip shield. DOE's assumed aqueous conditions are reasonable because titanium general corrosion proceeds more rapidly in aqueous conditions than in dry conditions. Further, the staff notes that data from DOE testing (SAR Figure 2.3.6-49) and independent corrosion data DOE referenced for material similar to the drip shield titanium alloys (Smailos and Köster, 1987aa) indicate that the general corrosion rates decrease for titanium alloys over time. The technical literature indicates that the decreasing corrosion rate may correspond to thickening of the passive oxide film (Jones, 1996aa). Thus, the NRC staff notes DOE's use of a constant corrosion rate over time is reasonable because this assumption will not underestimate the corrosion rate.

In addition, the NRC staff reviewed the DOE assumption that the general corrosion rate is independent of temperature. The studies DOE cited (e.g., Smailos and Köster, 1987aa; Hua and Gordon, 2004aa) considered materials analogous to the drip shield titanium alloys and environmental conditions that are similar to or more aggressive than those expected in the repository. On the basis of information provided in these studies, the NRC staff notes that, although the corrosion rates of titanium alloys may have some temperature dependence during an initial period of exposure to corrosive brines, there is little temperature dependence after this period. The information DOE provided is also consistent with independent analyses by NRC (Mintz and He, 2009aa), which confirmed that corrosion rates for titanium alloys do not show significant temperature dependence for temperatures that are representative of the repository conditions. Therefore, the NRC staff notes DOE's assumption that the corrosion rates of the drip shield titanium alloys are independent of temperature is reasonable because this assumption will not underestimate the corrosion rate.

Long-Term Corrosion Test Data

The corrosion rates for Titanium Grades 7 and 29 that were sampled in the Total System Performance Assessment (TSPA) were based on data from weight-loss corrosion tests at the Long-Term Corrosion Test Facility (SAR Section 2.3.6.8.1.2.1). The following summarizes the NRC staff's review of DOE's data implemented in the TSPA analysis.

Titanium Grade 7

The corrosion rate for Titanium Grade 7 that was sampled in the TSPA was based on 2.5-year tests of Titanium Grade 7 crevice and weight-loss specimens with wrought (base metal-type) and as-welded metallurgical conditions (SAR Section 2.3.6.8.1.2.1). Some specimens were fully immersed in solution (i.e., aqueous phase), whereas others were in the saturated vapor above the aqueous phase. DOE exposed the test specimens to different solutions, including simulated acidified water, simulated dilute water, and simulated concentrated water, the compositions of which are given in SAR Table 2.3.6-1. The tests were performed at temperatures of 60 and 90 °C [140 and 194 °F]. DOE measured the material weight loss during the test period and used these data to calculate the general corrosion rates, following American Society of Testing and Materials (ASTM) G1-90 (ASTM International, 1999aa). DOE observed that the corrosion rates of crevice specimens were lower than those of weight-loss specimens (SAR Figure 2.3.6-44). Therefore, DOE chose to use only the data from the weight-loss specimens in the model abstraction because it will calculate a higher corrosion rate in the TSPA analysis. For the weight-loss specimens, DOE did not observe a significant difference in corrosion rates between wrought and as-welded materials, but did observe that the corrosion rates depended upon the chemistry of the test solution. In particular, the corrosion rates for specimens tested in the simulated concentrated water aqueous phase were as high as 50 nm/yr [1.97×10^{-6} in/yr], whereas the corrosion rates for the specimens tested in the aqueous and vapor phases of simulated acidified water and simulated dilute water , as well as for specimens tested in simulated concentrated water vapor phase, were below 20 nm/yr [7.87×10^{-7} in/yr] as shown in BSC Figures 6.6[a] and 6.7[a] (2004as).

In the TSPA analysis, DOE assumed that corrosion occurs simultaneously on the inner surface and the outer surface of the Titanium Grade 7 drip shield plates, with different corrosion rates for the respective surfaces. DOE assumed the outer surface of the plate corroded faster than the inner surface because the outer surface is expected to be exposed to a more aggressive environment, including dust and dripping seepage water, as detailed in BSC Section 6.1.6[a] (2004as). DOE used the data from the most aggressive test condition, obtained from the simulated concentrated water aqueous phase, to derive the distribution from which the outer surface corrosion rate was sampled in the TSPA model. In aqueous simulated concentrated water, DOE measured higher corrosion rates for Titanium Grade 7 at 90 °C [194 °F] than at 60 °C [140 °F] as shown in BSC Figure 6.6[a] (2004as). DOE did not, however, consider temperature dependence for the titanium general corrosion rate. Instead, DOE elected to use only the data from the 90 °C [194 °F] tests because these gave a higher corrosion rate. These data (“Aggressive Condition” in SAR Figure 2.3.6-46) have a mean corrosion rate of 46.1 nm/yr [1.81×10^{-6} in/yr]. For the general corrosion rate on the underside of the drip shield plates, DOE used the data from specimens tested at 60 and 90 °C [140 and 194 °F] in the aqueous and vapor phases of the simulated acidified water and the simulated dilute water , respectively, as well as specimens tested at 60 and 90 °C [140 and 194 °F] in the simulated concentrated water vapor phase, as detailed in BSC Section 6.1.7(a) (2004as). These data (“Benign Condition” in SAR Figure 2.3.6-46) have a mean corrosion rate of 5.1 nm/yr [2.01×10^{-7} in/yr].

DOE considered uncertainty in the measured corrosion rates, which it attributed to difficulties in cleaning and weighing corrosion specimens, particularly given the very small weight losses associated with low corrosion rates, as well as randomness in the general corrosion processes, as described in BSC Section 6.1.6.1[a] (2004as). DOE determined that the corrosion rate for the outside of the drip shield plates is best represented by a normal distribution, the mean of which is sampled from a *t*-distribution, described in SNL Table 6.3.5-3 (2008ag). The *t*-distribution is a broader distribution DOE used given that this set of corrosion rate data only has six data points. The mean of the *t*-distribution is approximately 46.1 nm/yr [1.81×10^{-6} in/yr], with 2.5th and 97.5th percentile values of approximately 43.0 and 49.1 nm/yr [1.69×10^{-6} in/yr and 1.93×10^{-6} in/yr], respectively, as detailed in BSC Section 6.1.6.2[a] (2004as). The variability of distributions for the general corrosion rate on the outside of the drip shield plates was shown in BSC Figure 6-11[a] (2004as). For the inside of the drip shield plates, DOE determined that the general corrosion rate is best represented by a gamma distribution, the mean of which is sampled from a normal distribution, described in SNL Table 6.3.5-3 (2008ag). The gamma distribution is a continuous skewed distribution function used to describe the distribution of variables that are positive and unbound. The mean of the normal distribution is approximately 5.1 nm/yr [2.01×10^{-7} in/yr], with 2.5th and 97.5th percentile values of approximately 3.5 nm/yr and 6.8 nm/yr [1.38×10^{-7} in/yr and 2.68×10^{-7} in/yr], respectively, as outlined in BSC Section 6.1.7.2[a] (2004as). The variability distributions for the general corrosion rate on the inside of the drip shield plates were shown in BSC Figure 6-19[a] (2004as).

DOE compared the corrosion rate sampled in the TSPA code to independently reported corrosion rates for analogous alloys in environments similar to or more aggressive than those expected in Yucca Mountain, as detailed in SAR Section 2.3.6.8.1.5 and BSC Section 7.2.1[a] (2004as). DOE concluded that the TSPA-calculated corrosion rates are consistent with corrosion rates measured by Smailos and Köster (1987aa) for titanium plus 0.2 wt% palladium in the temperature range of 90 to 200 °C [194 to 392 °F] in a pH 4.9 chloride-sulfate brine.

In response to the NRC staff's request for additional information (RAI) on how the experimental uncertainties associated with sample cleaning, weighing, and measuring were incorporated into the sampled corrosion rate distributions, DOE (2009cn) stated that subsequent examination of corrosion test specimens revealed that posttest specimen cleaning did not adequately remove a residual oxide film. This resulted in under-measurements of specimen weight loss and, in turn, an underestimation of the general corrosion rates for the inside and outside of the drip shield plates. To assess the effect of the incomplete specimen cleaning procedure on corrosion rate uncertainties, DOE conducted cross section analyses of the chemically cleaned posttest specimens. DOE estimated that the general corrosion rates for Titanium Grade 7, presented in SAR Section 2.3.6.8.1, were underestimated by, at most, a factor of two. Consequently, DOE conducted a sensitivity analysis in which it considered corrosion rates up to four times those given in SAR Section 2.3.6.8.1. This shortened the drip shield framework and plate lifetime compared to those calculated in the TSPA model. DOE stated that this sensitivity analysis showed that corrosion rates of up to four times higher than those given in SAR Section 2.3.6.8.1 resulted in negligible differences in the expected dose curves, as shown in DOE Enclosure 5, Figure 2 (2009cn). Therefore, DOE concluded that the data presented in SAR Section 2.3.6.8.1 were reasonable to use in the TSPA model because unquantified experimental uncertainties had negligible impact on the postclosure performance assessment. Nevertheless, DOE stated it would update the SAR to incorporate the analysis provided in the response to DOE Enclosure 1 (2009cn).

In DOE Enclosure 4 (2009cn), DOE responded to the NRC staff's request for additional information seeking justification that the immersion test conditions in simulated brines to determine general corrosion rates are adequate to model the corrosion behavior of the drip shield, considering that some passive alloys may be more susceptible to corrosion in dripping conditions than in immersion conditions (e.g., Lee and Solomon, 2006aa). DOE stated that the temperatures at which dripping effects on corrosion behavior have been observed in other passive alloys are greater than the temperatures expected for the drip shield in dripping conditions. Moreover, DOE stated that data in the technical literature indicate that titanium alloys are highly resistant to dripping effects because of their tenacious passive film (Schutz, 2005aa). Therefore, DOE concluded that the immersion tests in the simulated brines were adequate to model the corrosion behavior of the drip shield in the repository because they accounted for potential dripping conditions.

Summary of NRC Staff's Review for Corrosion of Titanium Grade 7

The NRC staff reviewed the information DOE provided in SAR Section 2.3.6.8.1 and DOE (2009cn) and noted the following:

- With regard to the material conditions, DOE's testing of Titanium Grade 7 specimens with wrought and as-welded microstructures is reasonable. DOE's tests on the material microstructures are reasonable because they are representative of the microstructures expected for Titanium Grade 7 drip shield plates in the base metal and in the region of the weld, respectively, on the basis of the fabrication procedures set forth in BSC (2007bu).

- With respect to the corrosion test solutions, DOE tested the Titanium Grade 7 specimens in a range of corrosion test solutions, including simulated acidified water, simulated dilute water, and simulated concentrated water. The corrosion rate for the drip shield in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the drip shield. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the drip shield corrosion rate for the conditions expected in the repository. The NRC staff notes that DOE's corrosion test solutions are more chemically aggressive than waters expected to occur within the repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. Therefore, DOE's model of the drip shield corrosion rate based on tests in simulated acidified water, simulated concentrated water, and simulated dilute water is reasonable.

- With regard to the testing conditions, DOE used immersion corrosion tests to represent corrosion behavior, including potential dripping conditions in the drift. Literature DOE cited (Schutz, 2005aa) and other independent literature (He, et al., 2007aa) show that titanium alloys are highly resistant to dripping-induced corrosion effects at temperatures expected in the repository because of their strong tendency to passivity. Therefore, DOE reasonably performed corrosion tests in immersion because this will not underestimate the corrosion rate in dripping conditions.

In addition, the NRC staff evaluated DOE's experimental procedures for cleaning, weighing, and measuring the corrosion rates of the test specimens. DOE identified deficiencies with its specimen preparation and cleaning that initially led to unquantified experimental uncertainties in the general corrosion rate for Titanium Grade 7 reported in SAR Section 2.3.6.8.1. DOE addressed the deficiencies by demonstrating that the actual corrosion rates for the Titanium Grade 7 drip shield plates would not exceed a factor of two to four times the corrosion rates sampled in the Total System Performance Assessment (TSPA) analysis and that the higher corrosion rates resulted in negligible changes in calculated results. In the TSPA, the sum of the mean general corrosion rates for the inside and outside of the Titanium Grade 7 drip shield plates is approximately 51.2 nm/yr [2.06×10^{-6} in/yr]. Therefore, increasing the corrosion rate by a factor of two to four would increase the mean corrosion rate to approximately 100 to 200 nm/yr [3.93×10^{-6} to 7.87×10^{-6} in/yr]. At this range of corrosion rates, breach of the 15-mm [0.59-in]-thick Titanium Grade 7 plate by general corrosion would generally occur between 75,000 and 150,000 years after repository closure. This is well beyond 12,000 years after repository closure. Prior to 12,000 after repository closure, DOE calculated that the waste package outer barrier may undergo localized corrosion if contacted by seepage water (SAR Section 2.3.6.4). Therefore, DOE's demonstration is reasonable that unquantified experimental uncertainties in the general corrosion rate for Titanium Grade 7, reported in SAR Section 2.3.6.8.1, will have a negligible effect on the postclosure performance assessment results.

Finally, the NRC staff compared the Titanium Grade 7 corrosion rates sampled in the TSPA code to independently reported corrosion rates for the same or similar material. The corrosion rates sampled in the TSPA code are similar to those measured in independent testing that DOE cited (Smailos and Köster, 1987aa; Mattsson and Olefjord, 1990aa). Further, the NRC staff considered the report by Schutz (2005aa), which stated that the corrosion rates of titanium alloys are negligible in a variety of solutions, including seawater up to 260 °C [500 °F], 62 percent $CaCl_2$ at 150 °C [302 °F], boiling solutions of 10 percent and 30 percent $FeCl_3$, and boiling saturated $MgCl_2$ solution. Additionally, the NRC staff performed independent corrosion tests of Titanium Grade 7 in 1 M NaCl solution at 95 °C [203 °F] and measured a corrosion rate of approximately 87 nm/yr [3.43×10^{-6} in/yr] (Brossia, et al., 2001aa; Brossia and Cragnolino, 2004aa). The NRC staff determined that the titanium alloys tested in these studies are analogous to the drip shield titanium alloys and that the environmental conditions (e.g., temperature and brine chemistry) considered in these studies are chemically and thermally more aggressive than those expected for the postclosure period in the repository. The corrosion rates measured in these studies are similar to the Titanium Grade 7 drip shield plate corrosion rates the TSPA code calculated. Therefore, the NRC staff notes that independent reports in the technical literature provide support for the corrosion rate DOE calculated in the TSPA code.

Titanium Grade 29

The corrosion rate for Titanium Grade 29 that was applied in the TSPA analysis was based on 42-day weight-loss measurements of Titanium Grades 7 and 29 specimens in solutions that DOE stated were representative of seepage water and deliquescent brines expected in the repository [SAR Section 2.3.6.8.1.3 and BSC Section 6.2.1[a] (2004as)]. The compositions of the brines are given in BSC Table 6-7[a] (2004as). For tests at 120 and 150 °C [248 and 302 °F], DOE calculated the ratios of the corrosion rates of Titanium Grade 29 to those of Titanium Grade 7. For a given test environment, DOE calculated that the corrosion rate of Titanium Grade 29 could be a factor of one to seven times higher than that of Titanium Grade 7 (SAR Figure 2.3.6-48). From these data, DOE developed a discrete probability distribution function summarized in BSC Table 6-8[a] (2004as), which gave the ratio for the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7. To calculate the corrosion rate for the Titanium Grade 29 structural supports in the TSPA model, DOE sampled the ratio from this probability distribution function and multiplied the sampled ratio by the corrosion rate on the outside of the Titanium Grade 7 plate (i.e., under aggressive conditions).

In BSC Section 6.2[a] (2004as), DOE acknowledged that it did not have long-term general corrosion data for Titanium Grade 29. DOE stated, however, that the passive films for both Titanium Grade 7 and Grade 29 are likely to be predominantly titanium oxide. DOE also stated that data show that the passive behavior for the respective alloys is the same for the range of brines expected in the repository (Andresen and Kim, 2006aa). Therefore, DOE concluded that the corrosion processes for Titanium Grades 7 and 29 are similar and that comparing the corrosion rates of the respective alloys in short-term tests is an adequate basis for calculating the long-term corrosion rate for Titanium Grade 29.

DOE reanalyzed the comparative corrosion data (DOE, 2009cn). DOE determined that the weight loss for the respective alloys was measured by a weighing balance that had uncertainty larger than most of the measured weight change values. DOE concluded that it was unable to make a meaningful distinction between actual material weight loss and measurement uncertainty. Further, DOE stated that for the same tests, corrosion rates were also measured by electrochemical impedance spectroscopy and linear polarization resistance (Andresen and Kim, 2006aa), with negligible difference for the respective alloys. On the basis of this reanalysis, DOE determined that, because there was no measurable difference between the corrosion rates for the respective alloys in the 42-day tests, the corrosion rate ratio described in

SAR Section 2.3.6.8.1.3 was not needed. DOE decided to follow an alternative approach in which the corrosion rates for the Titanium Grade 29 structural supports are the same as the corrosion rate for the outer surface of the Titanium Grade 7 plate. DOE stated that this approach is justified because the corrosion rates measured by electrochemical impedance spectroscopy and linear polarization resistance for the respective alloys in the 42-day tests were nominally identical (Andresen and Kim, 2006aa). DOE also referenced Schutz (2005aa), which showed that the corrosion rates of Titanium Grades 7 and 29 are similar when exposed in a chloride solution with pH greater than 1, as shown in BSC Figure 6-22[a] (2004as).

In addition, DOE (2009cn) performed a sensitivity analysis using the TSPA model that compared the approach described in SAR Section 2.3.6.8.1.3 (in which the ratio for the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7 was sampled from a probability distribution function with a value in the range of approximately one to seven) to the new approach, in which the corrosion rate of Titanium Grade 29 is assumed to be equivalent to that of Titanium Grade 7. The analysis revealed that the drip shield structural framework failure time occurred later for the new approach, as shown in DOE Enclosure 2, Figure 1 (2009cn). The analysis also showed that, in the event of a seismic ground motion, the new approach gives a median dose that is about 25 percent higher between 80,000 and 300,000 years after repository closure, due to increased probability of waste package damage, as shown in DOE Enclosure 2, Figure 3 (2009cn). DOE stated that the mean expected dose was nearly the same for the respective approaches because the contribution of the seismic ground motion modeling case to the total mean annual dose is small during this time period. Therefore, DOE concluded that the data presented in SAR Section 2.3.6.8.1 were usable in the TSPA calculation, because unquantified experimental uncertainties had a negligible effect on the results from the postclosure performance assessment calculation. Nevertheless, DOE stated it would update the SAR to incorporate the analysis provided in DOE (2009cn).

Summary of NRC Staff's Review for Corrosion of Titanium Grade 29

The NRC staff reviewed the DOE approach to calculate the general corrosion rate of Titanium Grade 29. The DOE assumption of equivalent corrosion rates for Titanium Grades 7 and 29, respectively, is primarily based on corrosion testing of the alloys in simulated seepage water and deliquescent brines, including fluoride-bearing solutions. The corrosion rate for Titanium Grade 29 in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the drip shield. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the Titanium Grade 29 corrosion rate in repository conditions. The NRC staff notes that the DOE corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. As such, the solutions given in BSC Table 6-7[a] (2004as) may be used to calculate the general corrosion rate of Titanium Grade 29, because they are more chemically aggressive than waters expected to occur within the repository drifts. Additional support for the assumption of equivalent corrosion rates was provided by Schutz (2005aa), who showed the equivalent corrosion rate is observed even in acidic solutions. Finally, the NRC staff also observed nearly equivalent corrosion rates during testing of Titanium Grades 7 and 29 in 1 M NaCl and 4 M $MgCl_2$ solutions at elevated temperatures (Mintz and He, 2009aa). The NRC staff recognizes that the assumption of equivalent corrosion rates would extend the drip shield framework lifetime compared to that calculated in the TSPA model, where the corrosion rate of Titanium Grade 29 was up to a factor of seven times faster than that for Titanium Grade 7. The NRC staff notes that the DOE sensitivity analysis indicated that the TSPA calculations would not significantly differ when implementing the assumption of equivalent corrosion rates.

Abstraction and Integration

For the Nominal Modeling Case in the TSPA analysis, DOE implemented the model abstraction for general corrosion of the drip shield in the waste package and Drip Shield Degradation Submodel, as described in SNL Section 6.3.5.1 (2008ag), in which the drip shields were distributed in the five percolation subregions. The Waste Package and Drip Shield Degradation Submodel considers only general corrosion breach of the Titanium Grade 7 plates. DOE concluded that the drip shield would protect the waste package against seepage if the drip shield plates are intact, even if the drip shield supports collapsed and the sidewall buckled (SAR Section 2.3.4.5.3.1). For each realization, DOE sampled one general corrosion rate for the outside of the drip shield plates (under the aggressive condition) and one for the inside of the drip shield plates (under the benign condition) from the respective distributions given in SNL Table 6.3.5-3 (2008ag). The corrosion rates were applied to all drip shields, regardless of the percolation subregion, such that all drip shields in a given realization failed at the same time. The output of the Waste Package and Drip Shield Degradation Submodel was a fraction of that for drip shields in each percolation subregion breached by general corrosion as a function of time. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

SAR Figures 2.1-8 and 2.4-24 showed the distribution of calculated failure times for the Titanium Grade 7 drip shield plates in the Nominal Modeling Case, on the basis of the model described in SAR Section 2.3.6.8.1. DOE’s analyses calculated that most drip shield failures occur between 260,000 and 340,000 years after repository closure. DOE Enclosure 5, Figure 1 (2009cn) showed a modified distribution of failure times considering both a higher corrosion rate (based on additional uncertainties associated with specimen cleaning) and lower corrosion rate (based on potential decrease in corrosion rate over time). The modified distribution shows that most drip shield plate failures occur between 80,000 and 500,000 years after repository closure. In either case, there is negligible probability of drip shield plate breach by nominal processes within 12,000 years after repository closure, the time period during which DOE calculates that the waste package is susceptible to localized corrosion if contacted by seepage water.

For the Seismic Ground Motion Modeling Case in the TSPA analysis, DOE also implemented the Waste Package and Drip Shield Degradation Submodel to calculate the timing and magnitude of drip shield plate breach by general corrosion, as outlined in SNL Section 6.6.1 (2008ag). Both the Titanium Grades 7 and 29 corrosion rates are sampled in this modeling case. The Titanium Grade 7 corrosion rate was sampled in the same manner as in the Nominal Modeling Case. For Titanium Grade 29 structural supports, DOE sampled the ratio of the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7 once per realization from the discrete probability distribution function summarized in BSC Table 6-8[a] (2004as). The ratio was applied to all drip shields in a realization. SAR Figures 2.1-11 and 2.4-24 showed the distribution of failure times for the Titanium Grade 7 drip shield plates in the Seismic Ground Motion Modeling Case. Most plate failures occur between 100,000 and 300,000 years after repository closure. There is negligible probability of drip shield breach within 12,000 years after repository closure because the general corrosion rate of the Titanium Grade 7 drip shield plates is low and the likelihood of plate failure by a seismic event is negligible before that time period. For the Titanium Grade 29 structural supports, DOE calculated that most drip shield frameworks failed between 20,000 and 170,000 years after repository closure, using the model described in

SAR Section 2.3.6.8.1 and DOE Enclosure 2, Figure 1 (2009cn). For the alternative approach, in which DOE assumed equivalent corrosion rates for the structural supports and the plate, DOE calculated that most frameworks failed between about 80,000 and 170,000 years after repository closure, as shown in DOE Enclosure 2, Figure 1 (2009cn).

Summary of NRC Staff's Review for Abstraction and Integration of General Corrosion of the Drip Shield

The NRC staff reviewed the implementation and integration of the model abstraction for general corrosion of the drip shield used in the postclosure performance assessment calculation. The NRC staff notes that DOE has provided information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that DOE's use of the model abstraction was consistent with the design features of the drip shield, including materials of construction and dimensions given in SAR Section 1.3.4.7. Further, with respect to the general corrosion rates, DOE justified the ranges of these parameters and accounted for uncertainty in the model abstraction. Therefore, DOE's implementation of the drip shield general corrosion model abstraction in the TSPA code is reasonable because it would not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Staff's Summary of Conclusions for General Corrosion of the Drip Shield

The NRC staff reviewed the DOE model abstraction for general corrosion of the drip shield that was implemented in the TSPA code. DOE used appropriate experimental tests and other independent technical literature to provide support for the model abstraction. Therefore, DOE reasonably accounted for general corrosion of the drip shield in the TSPA code.

2.2.1.3.1.3.1.2 Drip Shield Early Failure

In SAR Section 2.3.6.8.4, DOE described how it developed the probability distribution for early failure of the drip shield that was sampled in the TSPA code. DOE assumed that a drip shield underwent early failure if it was emplaced in the repository with an undetected manufacturing- or handling-induced defect. On the basis of the processes associated with drip shield manufacturing and handling, DOE concluded that the probability of a drip shield early failure is best represented in the TSPA by a lognormal distribution with a mean of 2.21×10^{-6} per drip shield and an error factor of 14, as shown in SNL Table 6-7 (2007aa). The NRC staff reviewed the adequacy of this probability distribution in TER Section 2.2.1.2.2.4. The implementation of this probability distribution is addressed in this section.

Drip Shield Early Failure Conceptual Model

In DOE's conceptual model for early drip shield failure, a drip shield with an undetected manufacturing- or handling-induced defect completely fails (i.e., is removed as a barrier to the flow of water) at the time of repository closure (SAR Section 2.3.6.8.4.4.1). DOE selected this representation because there are uncertainties associated with the timing and extent of breach for defective drip shields and a completely degraded drip shield at the time of repository closure will not underestimate the timing and magnitude of radionuclide releases, as described in SNL Section 6.5.2 (2007aa). DOE concluded that this is a conservative representation of the early failed drip shield because the most likely consequence of improper drip shield manufacturing or handling would be stress corrosion cracking. DOE excluded drip shield stress corrosion cracking from the performance assessment because even if cracking occurred, the cracks would not affect the drip shield performance, because advective flow through cracks in the drip shield is also excluded from the performance assessment (SAR Section 2.3.6.8.3).

Summary of NRC Staff's Review of Drip Shield Early Failure Conceptual Model

The NRC staff reviewed DOE's conceptual model for drip shield early failure, as described in SAR Section 2.3.6.8.4. The NRC staff notes that DOE attributed no barrier capability to the early failed drip shield. However, consequences of manufacturing- or handling-induced defects (e.g., increased probability of stress corrosion cracking) would likely allow the drip shield to maintain some barrier capability, which limits radionuclide releases. Because early failed drip shields in the DOE model are assumed to have no barrier capability, the NRC staff notes that the model will not cause DOE to underestimate the timing or magnitude of radionuclide releases. Therefore, the DOE conceptual model drip shield early failure is reasonable.

Abstraction and Integration

The model abstraction for early failure of the drip shield was implemented in the TSPA calculation in the Drip Shield Early Failure Modeling Case, as described in SAR Section 2.4.2.1.5.2 and SNL Section 6.4.1 (2008ag). This modeling case uses most of the same modeling components and submodels as were implemented in the Nominal Modeling Case. In the Nominal Modeling Case, however, the Waste Package and Drip Shield Degradation Submodel calculates the waste package and drip shield breached areas as a function of time and passes this to the Engineered Barrier System Flow and Transport Submodels and the Waste Form Degradation and Mobilization Model Components. In the Drip Shield Early Failure Modeling Case, the Waste Package and Drip Shield Degradation Submodel was replaced with the drip shield early failure mode, which simulated early failure by removing a selected drip shield as a barrier to seepage at the time of repository closure.

In the Drip Shield Early Failure Modeling Case, the underlying waste package immediately experienced initiation of localized corrosion if the early failed drip shield was in seepage conditions. If the early failed drip shield was not in seepage conditions, the underlying waste package did not experience initiation of localized corrosion. In the TSPA model, DOE calculated the dose consequence of a drip shield early failure in each of the five percolation subregions for both commercial spent nuclear fuel (CSNF)-type and codisposal (CDSP)-type waste packages. DOE then calculated the expected dose using the early failure probability [sampled from the distribution given in SNL Table 7-1 (2007aa)], the distribution for the waste package type, and the seepage fraction for each percolation bin.

DOE calculated that there is approximately 98.3 percent probability of no drip shield early failures, approximately 1.6 percent probability of one drip shield early failure, and approximately 0.1 percent probability of two or more drip shield early failures, as shown in SNL Table 6.4-1 (2008ag). Using the TSPA model, DOE calculated that drip shield early failure has a negligibly small contribution to the calculated mean annual dose during the first 10,000 years following closure {less than 10^{-5} mSv [0.001 mrem]}, with a declining contribution thereafter (SAR Figure 2.4-18).

Summary of NRC Staff's Review for Abstraction and Integration of the Drip Shield Early Failure Model

The NRC staff reviewed the implementation of the drip shield early failure model in the TSPA calculation, as described in SNL Section 6.4.1 (2008ag). DOE has provided information for the

NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that the model abstraction was consistent with the design features of the drip shield, including materials of construction and dimensions given in SAR Section 1.3.4.7. Further, DOE justified the ranges of parameters and accounted for uncertainty in the model abstraction. Therefore, DOE's implementation of the drip shield early failure model abstraction in the TSPA analysis is reasonable because it will not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Summary of Conclusions for Drip Shield Early Failure

The NRC staff reviewed the DOE model abstraction for early failure of the drip shield that was implemented in the TSPA code. The NRC staff notes that DOE reasonably accounted for drip shield early failure in the Total System Performance Assessment (TSPA).

2.2.1.3.1.3.2 Waste Package Degradation

In SAR Section 1.5.2, DOE stated that waste packages are relied upon to limit water contacting the waste form and to prevent the mobilization of radionuclides. The waste package will have an outer barrier that is fabricated from a material that is expected to be corrosion resistant in the range of environmental conditions expected in the repository (SAR Section 2.3.6.1). In particular, the waste package outer barrier will be fabricated from Alloy 22 (UNS N06022), which is a nickel-chromium-molybdenum alloy.

In developing the postclosure performance assessment, DOE evaluated a number of features, events, and processes in SAR Table 2.2-5 related to chemical degradation of the waste package. These features, events, and processes include

- General corrosion of waste packages [features, events, and processes (FEP) 2.1.03.01.0A]
- Stress corrosion cracking of waste packages (FEP 2.1.03.02.0A)
- Localized corrosion of waste packages (FEP 2.1.03.03.0A)
- Hydride cracking of waste packages (FEP 2.1.03.04.0A)
- Microbially influenced corrosion (MIC) of waste packages (FEP 2.1.03.05.0A)
- Internal corrosion of waste packages prior to breach (FEP 2.1.03.06.0A)
- Early failure of waste packages (FEP 2.1.03.08.0A)
- Creep of metallic materials in waste packages (FEP 2.1.07.05.0A)
- Localized corrosion on waste packages outer surface due to deliquescence (FEP 2.1.09.28.0A)
- Thermal sensitization of waste packages (FEP 2.1.11.06.0A)

DOE included general corrosion, stress corrosion cracking, localized corrosion, MIC, and early failure in the postclosure performance assessment. The other features, events, and processes were screened from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these features, events, and processes from the performance assessment is found in TER Section 2.2.1.2.1.3.

In the TSPA analysis, DOE calculated that, due to its corrosion resistance, the waste package will significantly reduce the amount of water contacting the waste form for hundreds of thousands of years after repository closure (SAR Section 2.1.2.2.6). Because of the importance of the waste package in the postclosure performance assessment, the NRC staff reviewed the DOE model abstractions for waste package chemical degradation. In the context of these reviews, the NRC staff recognized that DOE attributed the high corrosion resistance of Alloy 22, in part, to the presence of its passive film. In the event of deterioration or loss of waste package passivity, the time to waste package breach may be sooner and the size of the breached area may be larger than DOE calculated in the TSPA code. As such, DOE stated that long-term persistence of the passive film on Alloy 22 is one of the key issues that it considered to determine the long-term performance of the waste package in the repository, as described in SNL Section 6.4.1.1 (2007al). In NRC Appendix D, Section 4.3.1 (2005aa), NRC also identified the long-term persistence of the passive film on the waste package outer barrier as being of high significance to risk for waste isolation.

Therefore, the following presents the NRC staff's first reviews DOE's approach to support its assessment of Alloy 22 passive film stability under the repository conditions. The NRC staff then conducts a detailed review of DOE's model abstractions for chemical degradation of the waste package, including general corrosion, microbially influenced corrosion (MIC), localized corrosion, stress corrosion cracking, and early failure.

Passivity of Alloy 22

In SAR Section 2.3.6.3.1 and SNL (2007al), DOE indicated that the stability of the Alloy 22 passive film depends primarily upon its physical and chemical properties, including microstructure, composition, and thickness. On Alloy 22 corrosion specimens, DOE investigated these passive film properties with various surface analytic techniques, including Auger electron spectroscopy, transmission electron microscopy, x-ray photoelectron spectroscopy, and electron energy loss spectroscopy (Orme, 2005aa). DOE performed short-term polarization tests, exposing Alloy 22 samples at 90 °C [194 °F] to solutions with a range of chemical compositions that DOE assumed were similar to, or more aggressive than, those expected in the repository (Orme, 2005aa). The solutions used in short-term polarization tests were either buffered 1 M NaCl solutions or multi-ionic solutions, including simulated acidified water, simulated concentrated water, and basic saturated water (compositions given in SAR Table 2.3.6-1). To assess the long-term passive film behavior, DOE examined 5-year U-bend samples of Alloy 22 exposed to simulated acidified water, simulated concentrated water, and simulated dilute water at 90 °C [194 °F] (Orme, 2005aa).

For both short- and long-term tests, DOE observed a thin, adherent passive oxide film on the surface of Alloy 22 corrosion specimens. The film typically had thickness in the range of 2 to 7 nm [7.87×10^{-8} to 2.76×10^{-7} in] and tended to be rich in chromium (III) oxides (Cr_2O_3 and/or $NiCr_2O_4$). In the solutions of acidic and near-neutral pH, a thick outer layer was also observed on the top of the inner chromium-rich oxide layer (Orme, 2005aa). The outer layer was porous and consisted mostly of nickel oxide and the oxides of some other alloying elements, including iron, tungsten, and molybdenum. In basic saturated water (pH ~12–13), DOE observed a thick silica deposit on Alloy 22 specimens (Orme, 2005aa), which DOE concluded arose from dissolution of test cell glassware or precipitation of silica from the test solution. In the case of 5-year U-bend samples exposed to simulated acidified water, simulated concentrated water, and simulated dilute water, all of the immersed samples had 100 to 5,000-nm [3.94×10^{-6} to 1.97×10^{-4}-in]-thick carbon and iron deposits on their surfaces. DOE determined the deposits are formed as leachates from either the walls of the test tanks or other metals in the tanks (Orme, 2005aa). Oil from the mill processing is also considered to be included in the deposits. DOE stated that, underneath these deposits, the passive film was still close to 5 nm [1.97×10^{-7} in] thick after 5 years' exposure. The presence of chromium-rich oxide passive film on the Alloy 22 surface was also observed at high temperatures {in the range of 120 to 220 °C [248 to 428 °F} in NaCl–$NaNO_3$–KNO_3 solutions (Orme, 2005aa; Dixit, et al., 2006aa).

To support the assessment of long-term passive film stability, DOE performed thermodynamic modeling with the EQ3/6 program (Orme, 2005aa). DOE concluded that this showed that chromium-rich oxides are stable on Alloy 22, which is consistent with empirical observation of passive film chemistry. Although the tests that DOE used to characterize the passive film of Alloy 22 were for a period of, at most, 5 years, DOE referenced the point defect film growth model, which states that the passive film on Alloy 22 will maintain steady-state thickness as the porous outer layer dissolves and the compact chromium-rich oxide inner layer is continuously regenerated.

After the SAR was submitted, DOE examined some 5- and 9.5-year Alloy 22 specimens from the Long-Term Corrosion Test Facility (SNL, 2009aa,ab). DOE identified thick organic deposits on some specimens. DOE responded to the NRC staff's request for additional information on evaluation of the effects of the carbon deposits on DOE's assessment of long-term passivity and corrosion behavior (DOE, 2009cl, 2010ae). In its response, DOE stated that the organic deposits observed on the Alloy 22 specimens most likely originated from lubricant or grease from mechanical equipment in the corrosion test facility (DOE, 2009cl, 2010ae). DOE did not identify evidence of either increased general corrosion rate or localized corrosion attack on the specimens. For these specimens, DOE measured a corrosion rate of 3 to 5 nm/yr [1.18×10^{-7} to 1.97×10^{-7} in] in simulated concentrated water at 60 °C [140 °F] after 9.5 years. DOE determined that this corrosion rate was consistent with that of uncontaminated specimens, as well as the waste package corrosion rate used in the Total System Performance Assessment (TSPA) model.

Therefore, DOE concluded that the organic deposits did not affect the assessment of long-term passivity or corrosion behavior.

Summary of NRC Staff's Review for Passivity of Alloy 22

The NRC staff reviewed the DOE approach to establish the stability of the waste package passive film in repository conditions. The NRC staff determined that DOE's assumption that the passive film will remain stable during the postclosure period is based, in part, on tests of Alloy 22 specimens in a range of corrosion test solutions, including simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water. Stability of the waste package passive film may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate in assessing waste package passive film stability in repository conditions. The NRC staff notes that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5 and waters considered in NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. Therefore, the DOE observations of Alloy 22 passive film stability in simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water support the DOE assumption of waste package passive film stability in repository conditions. Further, the NRC staff (Dunn, et al., 2005aa) and others (Lloyd, et al., 2003aa, 2004aa; Gray, et al., 2006aa; Montemor, et al., 2003aa; Hur and Park, 2006aa; Mintz and Devine, 2004aa) reported similar descriptions of the passive film for Alloy 22 and analogous nickel-based alloys (e.g., Alloy C-4, C276, 600, 625, and 690).

In regard to the organic deposits on the corrosion specimens, the NRC staff reviewed the DOE reports (SNL, 2009aa,ab) and determined that the presence of the deposits did not significantly affect the measured corrosion rates, given the range of uncertainty for such measurements, nor was there evidence of localized corrosion (e.g., pitting) on the specimens. The NRC staff notes that DOE conducted a reasonable analysis to show that organic deposits are not likely to affect anodic and cathodic reactions on the Alloy 22 surface. Therefore, the carbon deposits on some Alloy 22 specimens did not affect the assessment of the passive film stability.

The NRC staff also reviewed DOE's use of thermodynamic modeling and the point defect film growth model to support the assessment of the waste package passive film stability. The NRC staff conducted an independent analyses and obtained similar results as DOE for temperatures up to 180 °C [356 °F] (Pensado, et al., 2002aa; Jung, et al., 2008aa). With respect to application of the point defect model in DOE's assessment of passive film stability, DOE's test results (Orme, 2005aa) indicate that a passive film with steady-state thickness would provide a nearly constant corrosion rate over time at a particular temperature, potential, and pH. The NRC staff notes that the results from the Miserque, et al. (2006aa) study confirm the suitability of the DOE approach for using a point defect film growth model to support passive film stability.

Although the SAR contained information regarding waste package passive film stability in repository conditions, the NRC staff identified three primary technical issues and requested additional information from DOE. These issues involved passive film degradation by (i) anodic sulfur segregation, (ii) dripping seepage water, and (iii) silica deposits on the waste package. The NRC staff's reviews of these technical issues are presented next.

Effect of Anodic Sulfur Segregation on Passive Film Stability

By independent analyses and review of the technical literature, the NRC staff identified anodic sulfur segregation as a potential mechanism that could compromise the long-term stability of the passive film on the waste package outer barrier (NRC, 2005aa; U.S. Nuclear Waste Technical Review Board, 2001aa). Anodic sulfur segregation is a process that reduces the corrosion resistance of nickel and nickel-iron alloys by inhibiting the formation of the passive film (Marcus and Talah, 1989aa; Marcus, et al., 1988aa, 1984aa,ab, 1980aa). During anodic sulfur segregation, sulfur, which may be an impurity in Alloy 22, segregates to the metal-passive film interface because of selective dissolution of bulk metal elements such as nickel and iron. When the amount of sulfur at the metal-passive film interface reaches a critical concentration of about one atomic layer thickness, passive film breakdown has been observed (Marcus and Grimal, 1990aa). Assuming that 100 percent of the sulfur atoms in the alloy are retained at the metal-film interface, Marcus (2001aa) estimated that it would take about 900 years for the passive film of Alloy 22 to break down if the sulfur content in Alloy 22 is 5 weight parts per million (ppm) and the passive current density is 1 nA/cm^2 [6.45 nA/in^2].

In DOE Enclosure 5 (2009cl), DOE stated that the potential for anodic sulfur segregation would be mitigated by the presence of the alloying elements chromium and molybdenum in Alloy 22. In particular, molybdenum would bond with sulfur to form a molybdenum sulfide that dissolves under aqueous conditions, thus preventing a stable sulfur monolayer from forming at the alloy-passive film interface. Citing the work of NRC (Jung, et al., 2007aa), DOE also stated that chromium oxides are thermodynamically stable compared with sulfides. Thus the presence of chromium will promote passivation in spite of adsorbed sulfur.

Summary of NRC Staff's Review for the Effect of Anodic Sulfur Segregation on Passive Film Stability

The NRC staff reviewed the DOE assessment of anodic sulfur segregation of the waste package outer barrier. The effects of chromium and molybdenum on sulfur segregation were not considered in the time to passive film breakdown estimated in Marcus (2001aa). Further, the NRC staff determined that reports of passive film breakdown by anodic sulfur segregation are primarily associated with iron and iron-nickel alloys that lack the alloying elements molybdenum and chromium. The presence of the alloying elements molybdenum and chromium has been shown to prevent passive film breakdown by anodic sulfur segregation in materials similar to Alloy 22. Literature reports (Costa and Marcus, 1993aa; Marcus and Grimal, 1990aa) indicate that chromium in Ni-xCr-10Fe alloys (x = 8, 19, and 34 at%) counteracted the detrimental effects of sulfur and promoted alloy passivation. Further, it was reported that preadsorbed sulfur monolayers on the surface of Ni-2~6Mo and Fe-17Cr-14.5Ni-2.3Mo stainless steel, respectively, were removed in the form of soluble molybdenum sulfides during alloy dissolution (Marcus and Moscatelli, 1989aa; Elibiache and Marcus, 1992aa). Moreover, Alloy 625 (Ni-21.5Cr-9Mo-4Fe) was resistant to localized creviced corrosion in 1 M NaCl solution containing 0.01 M $Na_2S_2O_3$ up to 80 °C [176 °F], whereas Alloy 600 (Ni-15.5Cr-8Fe) with no molybdenum was attacked by crevice corrosion in the same solution at 20 °C [68 °F] (Mulford and Tromans, 1988aa).

The NRC staff also determined that the information DOE provided is consistent with independent investigations NRC conducted (Ahn, et al., 2008aa; Jung, et al., 2007aa, 2008aa). In particular, NRC performed electrochemically accelerated dissolution tests in aggressive solutions to assess whether sulfur would segregate to the surface of Alloy 22 during anodic dissolution. Surface analysis showed that almost all sulfur on the surface dissolved during the tests, such that a critical sulfur concentration for passive film breakdown was not reached. Scratch repassivation tests of Alloy 22 in sulfide-containing solutions showed that even if passive-film breakdown occurred, the material repassivated within a few seconds, which would limit the potential for corrosion degradation (Jung, et al., 2008aa). These experimental results also are supported by thermodynamic calculations that calculate a formation of stable chromium oxide (Cr_2O_3) and possible soluble molybdenum sulfide (MoS_2) in the range of temperatures, potentials, and pH expected for repository conditions (Jung, et al., 2007aa, 2008aa).

The NRC staff therefore notes that in the aerated oxic environment expected in the repository, sulfur that accumulates on the waste package surface will most likely be removed by dissolution in either the reduced form or as soluble molybdenum sulfides. These processes are expected to mitigate the potential for anodic sulfur segregation to affect the long-term stability of passive films on Alloy 22. The NRC staff notes that DOE has provided information to show that the potential effects of sulfur segregation do not need to be considered in the DOE model for passive film stability on the waste package outer barrier.

Dripping Effects on the Passive Film Stability

DOE used corrosion data from immersion experiments to develop the model abstraction for general corrosion of the waste package outer barrier (SAR Section 2.3.6.3.2). DOE identified the possibility, however, that conditions in the repository may lead to dripping seepage water contacting the waste package surface (SAR Section 2.1.2.2.6). The NRC staff determined that literature information in Ashida, et al. (2007aa, 2008aa) and Oka, et al. (2007aa) suggests that general corrosion processes may be affected if environmental conditions changed from immersion to dripping. Ashida, et al. (2008aa) observed salt deposit formation and localized corrosion (i.e., pitting and intergranular corrosion) on Alloy 22 specimens exposed to dripping of simulated concentrated water for 40 days at 90 °C [194 °F]. The micropits observed were, however, not stable, and there was no evidence for propagation of these micropits. Ashida, et al. (2007aa) also reported an increase of the passive current density of Alloy 22 due to dripping induced temperature fluctuations at 90 °C [194 °F].

In DOE Enclosure 1 (2009cm), DOE assessed the corrosion behavior of Alloy 22 under dripping conditions in the repository environment. DOE stated that the Alloy 22 sample tested in Ashida, et al. (2008aa) was thermally aged, resulting in a significant second phase precipitation. This precipitate can decrease a resistance to localized corrosion. DOE stated that the Alloy 22 for the waste package outer barrier will be solution annealed, eliminating the second phase precipitates. Therefore, DOE concluded the material Ashida, et al. (2008aa) evaluated was not relevant for the waste package. In the case of the change in passive current density due to temperature fluctuation as shown in the polarization test in Ashida, et al. (2007aa), DOE stated that the current TSPA model of general corrosion of Alloy 22 considers the change of general corrosion rate depending on the temperature (see TER Section 2.2.1.3.1.3.2.1); therefore, the increase of corrosion rate observed in Ashida, et al. (2007aa) is consistent with the TSPA general corrosion model.

Summary of NRC Staff's Review for Dripping Effects on the Passive Film Stability

The NRC staff reviewed the information provided in DOE Enclosure 1 (2009cm). The NRC staff notes that solution annealing of the waste package will eliminate the second phase particles that decreased localized corrosion resistance in the specimens tested by Ashida, et al. (2008aa). Nevertheless, the NRC staff determined that the second phase may not be eliminated in the region of the waste package closure weld, because this weld will not be solution annealed (SAR Section 1.5.2.7). Although Ashida, et al. (2008aa) observed micropits in such material conditions, the same report shows no evidence for stable pit propagation (Ashida, et al., 2008aa). Moreover, NRC's independent analyses (Dunn, et al., 2006ab) identified no localized corrosion (e.g., pitting or intergranular corrosion) of the mill-annealed Alloy 22 after dripping simulated pore waters onto the Alloy 22 specimens at 110 °C [230 °F] for 10 days. Therefore, micropitting of the waste package closure weld is unlikely to have a significant effect on the waste package barrier capability.

Further, the NRC staff noted that the temperature of the water used in the dripping tests in Ashida, et al. (2007aa) was close to room temperature. This resulted in a relatively large difference in the temperature between the room temperature of the dropped water and the hot surface of the Alloy 22 test specimen {i.e., 90 °C [194 °F]}, thereby contributing to the observed increase in passive current due to temperature fluctuation. Such temperature fluctuations will be much smaller for the waste package in the repository because of a relatively smaller difference in temperature between the waste package and the drift wall (Jung, 2010aa). The

NRC staff notes that this small temperature fluctuation is not likely to have a significant effect on the waste package general corrosion rate.

The NRC staff notes that DOE has provided information to show that dripping conditions in the repository will not affect the assessment of the waste package passive film stability.

Effect of Silica Deposits on Alloy 22 Passivity

DOE data and information in the technical literature indicate that silica deposits on Alloy 22 may affect the passive film. In basic simulated water (pH ~12 to 13), DOE observed a thick silica deposit on Alloy 22 specimens (Orme, 2005aa), which DOE concluded arose from dissolution of test cell glassware or precipitation of silica from the test solution. DOE also noted the presence of silica in salt deposits on Alloy 22 specimens exposed to simulated dilute water, simulated acidified water, and simulated concentrated water in the Long-Term Corrosion Test Facility as shown in Wong, et al., Table 4, Figures 2 and 3 (2004aa). In another experiment (Dixit, et al., 2006aa) DOE observed silica deposits on Alloy 22 specimens that experienced localized corrosion in a deaerated concentrated solution at 220 °C [428 °F]. Finally, information in Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) indicated that the presence of silica deposits can be associated with intergranular attack and stress corrosion cracking in nickel-based alloys in steam generator environments.

In DOE Enclosure 4 (2009cl) and DOE Enclosure 2 (2009cm), DOE stated that the presence of silicate did not significantly impact the corrosion potential and corrosion rate of Alloy 22 for tests conducted in simulated acidified water and NaCl solutions. DOE also presented experimental data (Andresen and Kim, 2007aa) for Alloy 22 tests in solutions of nitrate, chloride, and bicarbonate with 0.27 m silicate. The tests showed that the general corrosion rate of Alloy 22 was 3 to 4 nm/yr [1.18×10^{-7} to 1.57×10^{-7} in/yr] after 62 months' immersion at 95 °C [203 °F], which is close to the measured corrosion rate in solutions without silicate. Finally, DOE stated that Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) and Dixit, et al. (2006aa) considered more aggressive environmental conditions than those expected in the repository. Therefore, DOE concluded that the observations are not relevant to the waste package in the repository.

Summary of NRC Staff's Review for Effect of Silica Deposits on Alloy 22 Passivity

The NRC staff reviewed DOE's approach to assess the effects of silica deposits on the waste package passive film stability. On the basis of the information DOE provided (e.g., Andresen and Kim, 2007aa), the NRC staff notes that silica deposits do not adversely impact Alloy 22 passivity, because there is not a significant difference in measured corrosion rates for specimens with and without silica deposits. Moreover, Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) and Dixit, et al. (2006aa) considered more aggressive environmental conditions than those expected in the repository.

DOE has provided information in SAR Section 2.3.6.3.1 and in responses to the NRC staff's request for additional information to support the concept that the waste package passive film will be stable during the postclosure period.

2.2.1.3.1.3.2.1 General Corrosion of the Waste Package Outer Barrier

In SAR Section 2.3.6.3, DOE defined general corrosion of the waste package outer barrier as uniform thinning by electrochemical processes at its corrosion potential. General corrosion could lead to the release of radionuclides from the waste package if the waste package wall is breached. General corrosion thinning may also make the waste package more susceptible to degradation processes such as stress corrosion cracking (SAR Section 2.3.6.5) or impacts caused by seismic ground motion (SAR Section 2.3.4.5). This section of the TER includes the NRC staff's review of the DOE model abstraction for general corrosion of the waste package outer barrier.

Waste Package General Corrosion Conceptual Model

In DOE's conceptual model for general corrosion of the waste package outer barrier, general corrosion starts at the time of repository closure (SAR Section 2.3.6.2.2). DOE assumed aqueous conditions because wet conditions give higher corrosion rates than dry conditions (SAR Section 2.3.6.3). In DOE's model, the general corrosion rate is a function of the waste package temperature and, at a given temperature, it is assumed to be constant over time (SAR Section 2.3.6.3.1). DOE used an Arrhenius-type equation (SAR Eq. 2.3.6-3) to calculate the temperature-dependent general corrosion rate of the waste package outer barrier in the temperature range of 25 to 200 °C [77 to 392 °F]. DOE also considered using a decreasing general corrosion rate over time as an alternative conceptual model (SNL, 2007al) but concluded that this would calculate a longer time to waste package failure.

DOE also considered that microbial activity in the repository could affect the waste package corrosion behavior—a phenomenon called microbially influenced corrosion (MIC) (SAR Section 2.3.6.3.3.2). DOE stated that microorganisms can change the electrochemical reactions on the material surface and change the type or degree of corrosion compared to that which would be measured in the absence of microorganisms. For example, microbially influenced corrosion can enhance the general corrosion rate of Alloy 22. In the DOE conceptual model, the waste package outer barrier is subject to microbially influenced corrosion when the relative humidity is sufficiently high for microbial activities. The effect of microbially influenced corrosion on the general corrosion rate is quantified by a unitless scalar called the microbially influenced corrosion enhancement factor. If the relative humidity is sufficiently high, the general corrosion rate in the absence of the microorganisms (SAR Eq. 2.3.6-3) is multiplied by the microbially influenced corrosion enhancement factor to give the enhanced general corrosion rate (SAR Eq. 2.3.6-4).

Summary of NRC Staff's Review for Waste Package General Corrosion Conceptual Model

The NRC staff reviewed the DOE conceptual model for general corrosion of the waste package outer barrier. The NRC staff reviewed the DOE model assumption that the temperature dependence of the general corrosion rate can be quantified with the Arrhenius-type equation. Corrosion involves chemical and/or electrochemical reactions and the transport of reacting species and ions on the metal surface—a process known to be thermally activated (Fontana and Greene, 1978aa). Further, the Arrhenius relationship is commonly used to characterize the temperature dependence of thermally activated processes (ASM International, 1987aa) and has frequently been used to describe the temperature dependence of the general corrosion rate (e.g., Pensado, et al., 2002aa; Dunn, et al., 2005aa; Lloyd, et al., 2003aa; Hua and Gordon, 2004aa). As such, DOE's use of SAR Eq. 2.3.6-3 to derive at the temperature-dependent general corrosion rate for the waste package outer barrier is reasonable.

The NRC staff also reviewed the DOE model assumption that the corrosion rate is constant over time at a given temperature. DOE provided experimental data showing that the measured general corrosion rate of Alloy 22 decreases over time at a given temperature for experiments up to 5 years in duration (SAR Figure 2.3.6-13). This decrease of the corrosion rate with time is also observed in other Alloy 22 corrosion tests (Hua and Gordon, 2004aa; Evans, et al., 2005aa,ab), including independent tests NRC performed (Dunn, et al., 2005aa). The NRC staff also reviewed DOE's alternative conceptual model where the temperature was realistically decreased as a function of time. The NRC staff notes that this alternative model would result in a longer time to waste package failure compared to DOE's primary conceptual model. Therefore, the use of a constant corrosion rate in the TSPA calculation is reasonable because it would not underestimate the time to waste package breach by general corrosion.

Finally, the NRC staff reviewed the DOE model assumption that microbially influenced corrosion will occur above a relative humidity threshold value. The NRC staff determined that, although adequate water supply may be a critical requirement for microbial growth in the repository, other factors may limit microbial growth even if there is sufficient water. As detailed in SNL Section 6.4 (2004ab) and Amy, et al. (2002aa), DOE's conceptual model did not take credit for a number of these factors, including

- Waste package temperatures may be too high to support microbial growth
- The seepage water brine's ionic strength may be too high to support microbial growth
- Nutrient supplies may be inadequate to support microbial growth
- The oxic environment in the repository may inhibit microbially influenced corrosion caused by sulfate- or nitrate- reducing microbes

DOE's use of a threshold relative humidity value for the onset of microbially influenced corrosion is reasonable because this will not underestimate the probability of microbially influenced corrosion.

General Corrosion Rate by Long-Term Weight-Loss Measurements

In SAR Eq. 2.3.6-3, DOE established the general corrosion rate at the baseline temperature of 60 °C [140 °F] from 5-year weight-loss experiments (SAR Section 2.3.6.3.2.1). DOE performed corrosion tests in the Long-Term Corrosion Test Facility at 60 and 90 °C [140 and 194 °F], using Alloy 22 specimens with two different geometries: weight-loss specimens and crevice specimens. For both specimen types, tests were performed on specimens with different metallurgical conditions (i.e., mill annealed and as-welded) and in different corrosion test solutions, including simulated acidified water, simulated concentrated water, and simulated dilute water. After 5 years of exposure to the test solutions, every specimen was covered with surface deposits. Therefore, the posttest specimens were cleaned and descaled in accordance with ASTM G 1-90 (ASTM International, 1999aa). DOE stated that the cleaning methods used to remove the scale from the tested samples did not significantly affect untested control samples. Therefore, without correction of any possible mass loss from the replicate untested control foil sample, DOE determined the general corrosion rates of Alloy 22 based upon the formula defined in ASTM G 1-90.

DOE summarized the results of its corrosion tests in SNL Section 6.4.3.2 (2007al). DOE stated that there was no appreciable difference between the general corrosion rates for mill-annealed and as-welded specimens, as shown in SNL Figures 6-14 and

6-19 (2007al). However, the measured corrosion rates for crevice specimens were higher than those for weight-loss specimens, as shown in SNL Figure 6-22 (2007al). For the weight-loss specimens, DOE determined that the mean general corrosion rate was 3.15 nm/yr [1.24×10^{-7} in/yr], with the ±1 standard deviation of 2.71 nm/yr [1.07×10^{-7} in/yr]. For the crevice specimens, DOE calculated a mean general corrosion rate of 7.36 nm/yr [2.90×10^{-7} in/yr] with ±1 standard deviation of 4.93 nm/yr [1.94×10^{-7} in/yr]. Because the crevice specimens tend to give higher corrosion rates, DOE only used the crevice data to develop the distribution from which the 60 °C [140 °F] general corrosion rate parameter was sampled in the TSPA code.

DOE determined that uncertainty and variability in the measured corrosion rate could be attributed both to measurement uncertainty, given the very small weight loss associated with low corrosion rates, and to actual variation in the corrosion processes on the material surface, as outlined in SNL Section 6.4.3.3 (2007al). In the model abstraction, DOE accounted for this uncertainty by fitting the 5-year corrosion data to the Weibull cumulative distribution functions, which are sampled in the Total System Performance Assessment (TSPA) code, as detailed in SNL Section 6.4.3.3.2 (2007al). DOE stated that the Weibull distribution was determined to be the best fit to the experimental data as compared to other fits such as uniform distribution, normal distribution, lognormal distribution, and gamma distribution. DOE characterized the Weibull distribution with two parameters: the scale factor and the shape factor. DOE used three different scale factor/shape factor pairs, corresponding to low, medium, and high uncertainty levels, to define three different Weibull distributions for the 60 °C [140 °F] general corrosion rate parameter, as shown in SNL Table 6-7 (2007al). In the TSPA code, the low, medium, and high general corrosion rate distributions were sampled such that the low and high distributions were each used for 5 percent of the realizations and the medium distribution was used for 90 percent of the realizations. DOE used the 5–90–5 percent partitioning to ensure that there were sufficient differences in the general corrosion rates for the respective distributions, yet that each distribution was amply sampled to produce a meaningful contribution (SAR Section 2.3.6.3.3.1). The Weibull distributions from which the 60 °C [140 °F] general corrosion rate of the waste package outer barrier is sampled in the TSPA code are shown in SAR Figure 2.3.6-9.

In response to the NRC staff's request for additional information that requested justification for the representation of uncertainties associated with cleaning the long-term corrosion specimens, DOE responded in DOE Enclosure 3 (2009cl) that the specimens were not adequately cleaned prior to performing weight-loss measurements. In particular, DOE determined that the initial weight of the specimens was artificially high because of the failure to remove mill-annealed oxide and surface contamination. This oxide and surface contamination, however, were removed during posttest cleaning. Nevertheless, DOE assumed that the associated weight loss was attributable to the general corrosion of Alloy 22. Thus, DOE concluded that it overestimated the actual weight loss of Alloy 22 and, in turn, overestimated the general corrosion rate. DOE stated that it is in the process of recleaning and reanalyzing the specimens following the procedures in ASTM G 1-03 (ASTM International, 2003ab). DOE provided preliminary data from the reanalysis for the weight-loss specimens in DOE Enclosure 3 (2009cl) and stated that those data gave the most accurate estimate for the general corrosion rate of Alloy 22. As shown in DOE Enclosure 3, Figure 8 (2009cl), the corrosion rate from the recleaned weight-loss specimens is close to or lower than that calculated by the three Weibull distributions DOE used in the TSPA code for the Alloy 22 general corrosion rate—particularly for corrosion rates with high cumulative probabilities. DOE stated that only corrosion rates with cumulative probabilities of 0.96 and above {corrosion rate greater than ~15 nm/yr [5.91×10^{-7} in/yr]} are important for waste package performance. Because the data from the recleaned, reanalyzed specimens provide lower corrosion rates than those calculated by the Weibull distributions at the high cumulative probabilities, DOE concluded that use of the Weibull distributions shown in SAR Figure 2.3.6-9 are reasonable because the waste package failure time was not overestimated. Nevertheless, DOE stated that the final recleaning and reanalysis of the corrosion specimens will be completed and documented in an update to the SAR.

Summary of NRC Staff's Review for General Corrosion Rate by Long-Term Weight-Loss Measurements

The NRC staff reviewed the DOE approach to establish the waste package general corrosion rate by long-term tests. With regard to the material conditions for the corrosion tests, DOE tested Alloy 22 specimens with mill-annealed and as-welded microstructures. The NRC staff notes that DOE performed tests on materials with these microstructures because they are representative of those expected for the waste package base metal and weld region, respectively, on the basis of the fabrication procedures set forth in SAR Section 1.5.2.7. With regard to the test solutions for the corrosion tests, DOE tested Alloy 22 specimens in a range of solutions, including simulated acidified water, simulated concentrated water, and simulated dilute water. The corrosion rate for the waste package in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the waste package corrosion rate in repository conditions. The NRC staff notes that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. Therefore, it was reasonable for DOE to model the waste package corrosion rate on the basis of tests in these simulated brines.

The NRC staff also notes that DOE adequately identified deficiencies with specimen preparation and cleaning that led to unquantified experimental uncertainties in the general corrosion rate for the waste package outer barrier reported in SAR Section 2.3.6.3. Further, the methodology DOE used for recleaning and reanalyzing weight-loss specimens is reasonable for such measurements and follows the standards specified in ASTM G–1-03 (ASTM International, 2003ab). Therefore, the recleaned, reanalyzed weight-loss data presented in DOE Enclosure 3 (2009cl) are reasonable to represent the general corrosion rate of Alloy 22 at 60 °C [140 °F].

The NRC staff compared the general corrosion rates measured from the recleaned, reanalyzed weight-loss specimens to the general corrosion rates the Weibull distributions calculated in the TSPA code. The NRC staff notes that, at lower cumulative probabilities, the corrosion rate calculated from the recleaned, reanalyzed weight-loss specimens may be slightly higher {by less than approximately 3 nm/yr [1.18×10^{-7} in/yr]} than the rate calculated from the Weibull distributions. The NRC staff determined, however, that because the corrosion rates are very low at this cumulative probability {less than approximately 7 nm/yr [2.76×10^{-7} in/yr]}, a difference of less than 3 nm/yr [1.18×10^{-7} in/yr] will have little effect on the waste package performance during the postclosure period. The NRC staff recognizes that, at higher cumulative probabilities (greater than approximately 0.8), the corrosion rate calculated from the recleaned, reanalyzed weight-loss specimens is less than the corrosion rates calculated from the Weibull distributions, even at the low uncertainty level. The NRC staff notes that the corrosion rates at the high end of the cumulative distribution, where the corrosion rate may be 10 nm/yr [3.94×10^{-7}in/yr] or higher, are most important for waste package performance. For corrosion rates at this level, the waste package may fail by general corrosion breach during the 1-million-year postclosure period or may be susceptible to damage in the event of seismic ground motion. Therefore, the NRC staff notes that, for corrosion rates most significant to waste package performance, the Weibull distributions DOE used to sample the 60 °C [140 °F] general corrosion rate in the TSPA calculate a higher corrosion rate than the most accurate experimental measures of the general corrosion rate. Further, sampling from the low, medium, and high Weibull distributions in 5, 90, and 5 percent of the realizations, respectively, is a reasonable approach to represent uncertainty in the general corrosion rate. The NRC staff notes that greater partitioning between the distributions (e.g., 1–98–1 percent) would give too few samples from the low and high distributions to make a statistically meaningful contribution to the cumulative distribution, whereas smaller partitioning between the distributions (e.g., 30–40–30 percent) would not give a statistically meaningful distinction between the respective distributions. On the basis of this information, the Weibull distributions from which DOE sampled the 60 °C [140 °F] general corrosion rate in the TSPA analysis are reasonable because they will not underestimate the general corrosion rates at high cumulative probabilities, which are most important for waste package performance.

Temperature Dependence of the General Corrosion Rate

DOE conducted experiments to determine the temperature dependence of the general corrosion rate of the waste package outer barrier by measuring the activation energy for general corrosion of Alloy 22 (SAR Section 2.3.6.3.2.2). DOE used the short-term electrochemical polarization resistance technique following the ASTM G 59-97 (ASTM International, 1998aa). Mill-annealed and welded specimens were tested in a range of solutions containing NaCl and KNO_3 at temperatures ranging from 60 to 100 °C [140 to 212 °F] (SAR Table 2.3.6-4). DOE used these solutions because they simulate the conditions of moderate relative humidity where calcium is expected to be a minor component in the aqueous environment in the repository, as outlined in SNL Section 6.4.3.4 (2007al).

From these data (SAR Figure 2.3.6-7), DOE used a linear mixed-effects statistical analysis to calculate a mean activation energy of 40.78 kJ/mol [9.74 kcal/mol], with a standard deviation 11.75 kJ/mol [2.81 kcal/mol]. DOE selected a normal distribution to represent the temperature-dependence term on the basis of statistical fitting techniques. The activation energies for the individual solutions used to determine the distribution of the activation energy are shown in SAR Table 2.3.6-5. DOE confirmed the activation energy calculated from these short-term polarization tests by comparisons to the activation energy from the long-term 5-year weight-loss data of Alloy 22 specimens immersed in simulated concentrated water at 60 and 90 °C [140 and 194 °F], respectively, as described in SNL Section 6.4.3.4 (2007al). DOE calculated a mean activation energy of 40.51 kJ/mol [9.68 kcal/mol] for the 5-year corrosion data, which is close to the mean calculated from the short-term polarization technique. From the 5-year corrosion data, the activation energy distribution was also obtained. This distribution was best represented by truncating the normal distribution of the short-term polarization tests at −3 and +2 standard deviations.

The deficiencies with cleaning and weighing Alloy 22 corrosion specimens discussed in DOE Enclosure 3 (2009cl), however, led DOE to reevaluate the calculation of the temperature dependence of the general corrosion rate. The deficiencies were not associated with the short-term polarization data, but rather the comparison of 5-year general corrosion rates for specimens immersed in simulated concentrated water. For the latter, DOE recalculated the activation energy using the corrosion rates measured for the recleaned, reanalyzed weight-loss specimens. From these data, DOE calculated a mean activation energy of approximately

32.26 kJ/mol [7.71 kcal/mol], with minimum and maximum values of 3.37 and 60.05 kJ/mol [0.81 and 14.3 kcal/mol], respectively, as shown in DOE Enclosure 3, Figure 9 (2009cl). These values are approximately 20 percent lower than the activation energies sampled from the truncated normal distribution described in SAR Section 2.3.6.3, which was sampled in the TSPA code. Using both the updated distribution for the activation energy, as shown in DOE Enclosure 3, Figure 9 (2009cl), and the updated distribution for the 60 °C [140 °F] general corrosion rate, as shown in DOE Enclosure 3, Figure 8 (2009cl), DOE calculated the temperature-dependent general corrosion rate of Alloy 22, using SAR Eq. 2.3.6-3. DOE compared the corrosion rates calculated using the updated distributions to the corrosion rates calculated using the model described in SAR Section 2.3.6.3. As shown in DOE Enclosure 3, Figures 10–12 (2009cl), for the temperature range of 25 to 200 °C [77 to 392 °F], the updated distributions derived from recleaned, reanalyzed weight-loss specimens give lower corrosion rates than obtained by the model described in SAR Section 2.3.6.3, which was implemented in the TSPA code. As such, DOE concluded that the TSPA code did not underestimate the waste package general corrosion rate.

Summary of NRC Staff's Review for Temperature Dependence of the General Corrosion Rate

The NRC staff reviewed the DOE approach to calculate the temperature dependence of the waste package general corrosion rate. The NRC staff reviewed the material and environmental conditions at which DOE measured the activation energy for general corrosion of the waste package outer barrier that was sampled in the TSPA code. With regard to the material conditions for the tests, DOE tested Alloy 22 specimens with mill-annealed and as-welded microstructures. The NRC staff notes that it was reasonable for DOE to perform tests on materials with these microstructures because they are representative of those expected for the waste package base metal and weld region, respectively, on the basis of the fabrication procedures set forth in SAR Section 1.5.2.7. With regard to the test solutions for the corrosion tests, the corrosion behavior of the waste package in the repository may depend upon such seepage water characteristics as the pH and concentration of ionic species. The corrosion rate for the waste package in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions described in SAR Section 2.3.6.2.2 to determine whether they are adequate to calculate the waste package general corrosion rate activation energy in repository conditions. The NRC staff notes that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5 and waters considered in NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. As such, the NRC staff notes that it was reasonable for DOE to calculate the activation energy for general corrosion of the waste package on the basis of tests in the brines described in SAR Section 2.3.6.2.2. The NRC staff also reviewed the experimental methodology used to measure the activation energy that was sampled in the TSPA code. The short-term polarization test used to measure the activation energy was reasonable for such measurements and conformed with ASTM G–59-97 (ASTM International, 1998aa).

Additionally, the NRC staff compared the activation energy for general corrosion of the waste package outer barrier calculated from the recleaned, reanalyzed weight-loss specimens to the activation energy calculated by the short-term polarization test. Both calculations, with means of 40.78 kJ/mol [9.74 kcal/mol] and 32.26 kJ/mol [7.71 kcal/mol], respectively, give an activation energy in the range that is independently reported in the technical literature, which is between approximately 25 and 55 kJ/mol [5.97 and 11.9 kcal/mol] (Lloyd, et al., 2003aa; Scully, et al., 2001aa; Hua and Gordon, 2004aa; Smailos and Köster, 1987aa). In particular, the independent testing NRC previously conducted gives an activation energy of approximately 45 kJ/mol [10.7 kcal/mol] (Dunn, et al., 2005aa; Pensado, et al., 2002aa). This activation energy is close to the mean value of 40.78 kJ/mol [9.74 kcal/mol] used in the TSPA code. Therefore, DOE reasonably identified and analyzed the uncertainties in activation energy introduced by deficiencies in DOE's weighing and measuring the 5-year weight-loss specimens. The staff also notes that the application of the normal distribution with truncation in the TSPA code is reasonable to represent the range of the activation energy.

Furthermore, in regard to DOE's calculated temperature-dependent general corrosion rates used in the TSPA model, the NRC staff assessed DOE's calculated corrosion rates by comparing them to the rates reported in independent studies in the technical literature. At 25 °C [77 °F], DOE's model calculates that the corrosion rate is less than 7 nm/yr [2.76×10^{-7} in/yr] (SAR Figure 2.3.6-11). This corrosion rate is lower than the rates measured at room temperature in other studies (McMillion, et al., 2005aa; Dunn, et al., 2005aa), which range from about 7 to 137 nm/yr [2.76×10^{-7} to 5.39×10^{-6} in/yr]. However, the NRC staff notes that the data of McMillion, et al. (2005aa) and Dunn, et al. (2005aa) are from short-term tests, whereas long-term experimental results show that the general corrosion rate of Alloy 22 decreases by up to two orders of magnitude as experiments progress beyond a few years. Therefore, the corrosion rates reported by McMillion, et al. (2005aa) and Dunn, et al. (2005aa) were overestimated in terms of long-term general corrosion rate by as much as two orders of magnitude at room temperature. The corrosion rate DOE calculated for 25 °C [77 °F] is not underestimated, and except as noted, is consistent with the corrosion rate in the literature (McMillion, et al., 2005aa; Dunn, et al., 2005aa). The NRC staff notes that DOE's corrosion rates at elevated temperatures of 150 and 200 °C [302 and 392 °F] (SAR Figure 2.3.6-11) are similar to, or greater than, those NRC measured (Yang, et al., 2007aa) for Alloy 22 at the same temperature range. The NRC staff also notes that these corrosion rates are greater than the corrosion rate of 120 nm/year [4.72×10^{-6} in/yr] that Smailos (1993aa) reported for Alloy C-4 (an analogue for Alloy 22) after 6 months' exposure to NaCl-rich brines at 150 °C [302 °F]. On the basis of this information, the distribution from which DOE samples the temperature dependence of the waste package outer barrier general corrosion rate in the TSPA model is reasonable because it is unlikely that DOE underestimated the corrosion rate for the range of temperatures that may be reasonably expected in the repository.

Microbially Influenced Corrosion Effects

The NRC staff reviewed DOE's approach to establish the conditions for the onset of microbially influenced corrosion and to quantify the extent to which microbially influenced corrosion may affect general corrosion behavior.

Threshold Relative Humidity for the Onset of Microbially Influenced Corrosion

As discussed in SAR Section 2.3.6.3.3.2 and DOE Enclosure 10 (2009cl), DOE concluded that the relative humidity in the repository must be greater than a threshold value for microbially influenced corrosion to increase the general corrosion rate of the waste package outer barrier. DOE used experimental data and information from the independent studies to determine this threshold relative humidity. DOE performed experiments in which Alloy 22 specimens were embedded in crushed Yucca Mountain tuff and placed in chambers with Yucca Mountain native microorganisms at different temperature and humidity levels (Else, et al., 2003aa). DOE reported that the optimum condition for microbial growth is at a temperature of 30 °C [86 °F] and

100 percent relative humidity. Microbial growth was extremely limited at higher temperatures or lower humidity. In SNL Section 6.4 (2004ab), DOE also cited several studies which indicate that robust growth of most microorganisms requires a relative humidity of 90 percent or higher, although limited growth is seen at relative humidity as low as 75 percent (e.g., Brown, 1976aa; Pedersen and Karlsson, 1995aa). In the TSPA code, DOE accounts for uncertainty in the threshold relative humidity by sampling this threshold relative humidity from a uniform distribution between 75 and 90 percent (SAR Section 2.3.6.3.3.2).

Summary of NRC Staff's Review for Threshold Relative Humidity for the Onset of Microbially Influenced Corrosion

The NRC staff reviewed the DOE approach to establish the threshold relative humidity for the onset of microbially influenced corrosion. DOE used experimental data and information from independent studies to establish the threshold relative humidity for microbial growth in conditions similar to Yucca Mountain. Moreover, the NRC staff notes that DOE did not credit additional factors that may preclude microbial growth in the repository even if the relative humidity exceeded the threshold value. Therefore, DOE's distribution for the threshold relative humidity is reasonable because it will not underestimate the probability of the onset of microbially influenced corrosion.

Microbially Influenced Corrosion (MIC) Factor

If the relative humidity at the waste package surface is greater than the threshold value, the microbially influenced corrosion-enhanced general corrosion rate for the waste package outer barrier is calculated by multiplying the general corrosion rate in the absence of the microorganisms (given by SAR Eq. 2.3.6-3) by the MIC enhancement factor. DOE performed laboratory tests to determine the extent to which microbially influenced corrosion may affect the general corrosion rate of Alloy 22 (SAR Section 2.3.6.3.2.3). DOE used the electrochemical polarization technique to measure the corrosion rate of Alloy 22 specimens in nutrient-enriched, simulated Yucca Mountain well water, with and without the presence of microbes (Lian, et al., 1999aa). Test results were shown in SNL Table 6-16 and Figure 6-54 (2007al). DOE concluded that the general corrosion rate for Alloy 22 in the microbial-rich solution was up to a factor of approximately two higher than the general corrosion rate in sterile solution. DOE represented epistemic uncertainty in the microbially influenced corrosion enhancement factor to account for natural variation in the expected extent of microbial activity in repository conditions. Thus, in TSPA code, DOE samples the microbially influenced corrosion enhancement factor from a uniform distribution between one (i.e., microbially influenced corrosion has no effect on the general corrosion rate) and two (i.e., general corrosion rate would be double the nominal general corrosion rate).

Summary of NRC Staff's Review for Microbially Influenced Corrosion (MIC) Factor

The NRC staff reviewed the DOE approach to calculate the microbially influenced corrosion enhancement factor. The NRC staff notes that the corrosion tests DOE performed to measure the microbially influenced corrosion enhancement factor were performed in conditions that would support the growth of microbes. Further, the NRC staff determined that the polarization tests used to measure the microbially influenced corrosion effect conformed to ASTM G–59-97 (ASTM International, 1998aa), which is appropriate for such measurements. The NRC staff also notes that DOE did not use sterile conditions in the long-term (5 years) corrosion tests used to determine the nominal general corrosion rate for the waste package outer barrier. DOE indicated that some samples from these tests contain a significant amount of microbial bacteria, even though no bacteria were deliberately introduced (Horn, et al., 2005aa). Therefore, DOE has selected a reasonable range to represent the microbially influenced corrosion enhancement factor in the TSPA analysis because it is unlikely to underestimate the extent to which Yucca Mountain microorganisms may increase the general corrosion rate of the waste package outer barrier.

Abstraction and Integration

The model abstraction for general corrosion of the waste package outer barrier is implemented in the Waste Package and Drip Shield Degradation Submodel in the Total System Performance Assessment (TSPA) code in SNL Section 6.5.3 (2008ag). The inputs that are needed for the model abstraction are temperature of the waste package and the relative humidity in the drift. These inputs come from the Engineered Barrier System Thermohydrologic Environment Submodel. The Waste Package and Drip Shield Degradation Submodel includes both the commercial spent nuclear fuel configuration that uses the transportation, aging, and disposal canister configuration parameters and the codisposal waste package configuration that uses the five high-level waste/one DOE spent nuclear fuel long configuration parameters. DOE assumed a total of 11,629 waste packages divided into 5 percolation subregions, each of which is subject to different environmental conditions, as detailed in SNL Section 6.3.5.1.3 (2008ag). The Waste

Package and Drip Shield Degradation Submodel general corrosion calculations are performed for both waste package configurations in each of the percolation subregions.

In the Waste Package and Drip Shield Degradation Submodel, the waste package surface is divided into subareas, referred to as patches, to account for the spatial variability of general corrosion on the waste package surface, as outlined in SNL Section 6.3.5.1.2 (2008ag). Each patch may have a different general corrosion rate. The submodel uses a patch area of 231.5 cm^2 [35.88 in^2], so the commercial spent nuclear fuel and the codisposal waste packages have 1,430 and 1,408 patches, respectively. For each realization, each patch is assigned a different value for the 60 °C [140 °F] general corrosion rate, which is sampled from the Weibull distributions derived from 5-year weight-loss corrosion data from creviced Alloy 22 specimens (SAR Section 2.3.6.3.3.1). Because the size of the crevice specimens that DOE used to measure the 5-year general corrosion was about one-fourth the patch size, DOE sampled the 60 °C [140 °F] general corrosion rate four times for each patch and applied the highest of the four sampled rates to the patch, as described in SNL Section 6.2.5.1.2 (2008ag). The effect of rescaling the 60 °C [140 °F] general corrosion rate distribution, shown in SNL Figure 6.3.5-6 (2008ag), resulted in rates that are approximately twice those of the nominal distribution. To account for the temperature dependence of the general corrosion rate, a single value of the temperature-dependence parameter is sampled in each realization from the distribution derived from short-term polarization tests and applied to all waste package patches. To account for potential microbially influenced corrosion, the value of the threshold relative humidity for microbially influenced corrosion is sampled once per realization from a uniform distribution in the range of 75 to 90 percent and applied to all waste packages. If the relative humidity in the drift exceeds the threshold, the microbially influenced corrosion enhancement factor is sampled from a uniform distribution in the range of one to two and applied to the patches.

DOE considered a waste package outer barrier to be breached by general corrosion when one or more patches are penetrated. When the waste package was breached, the general corrosion model was also applied to the inner surface of the waste package outer barrier. The output for the general corrosion model gave the percentage of breached waste packages as a function of time and the average number of patch penetrations per breached waste package as a function of time. This output was transferred to and used in the waste form degradation, and the mobilization model component and the engineered barrier system flow and engineered barrier system transport submodels. SAR Figures 2.1-10(b) and 2.1-16(b) showed the fraction of commercial spent nuclear fuel waste packages breached by general corrosion and the fraction of the waste package surface area breached per breached waste package, respectively, for the commercial spent nuclear fuel waste package in the Nominal Modeling Case.

A mean of less than 10 percent of commercial spent nuclear fuel waste packages are breached over 1 million years, and of the breached waste packages, the mean breached area is less than 0.3 percent of the total waste package surface area. The results for the codisposal waste package in the Nominal Modeling Case are similar [SAR Figure 2.1-17(b)]. DOE Enclosure 1, Figures 9 and 10 (2009bj) showed the fraction of commercial spent nuclear fuel and codisposal waste packages, respectively, breached by general corrosion in the Seismic Ground Motion Modeling Case. For both waste packages, the mean is approximately 10 percent breached in 1 million years. DOE Enclosure 1, Figures 11 and 12 of its response to the NRC staff's request for additional information (DOE, 2009bj) showed the fraction of the surface area breached for the commercial spent nuclear fuel and the codisposal waste packages breached by general corrosion in the Seismic Ground Motion Modeling Case. For both waste packages, the fraction is approximately 1 percent of the surface area.

In regard to the activation energy for general corrosion in TSPA model, the NRC staff notes that DOE performed sensitivity analyses which show that the expected dose has a strong correlation to the activation energy for general corrosion of the waste package outer barrier and tends to increase with decreasing activation energy for general corrosion (SAR Figures 2.4-151 and 2.4-155).

Summary of NRC Staff's Review for Abstraction and Integration for General Corrosion of the Waste Package Outer Barrier

The NRC staff reviewed the implementation and integration of the model abstraction for general corrosion of the waste package outer barrier in the postclosure performance assessment. DOE has provided information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determines that the model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the NRC staff notes that, with respect to the general corrosion rate, DOE adequately justified the ranges of these parameters and accounted for uncertainty in the model abstraction. Moreover, DOE reasonably accounted for spatial variability in the general corrosion rate on the waste package surface by applying the corrosion rate on a patch scale and by rescaling the corrosion rate to account for the difference between the patch size and size of the corrosion specimens. The NRC staff performed independent calculations to confirm the Waste Package and Drip Shield Degradation Submodel in the TSPA (Jung, 2010aa). The NRC staff's calculations, with respect to the timing and magnitude of waste package breach by general corrosion, were consistent with DOE's calculations. Therefore, DOE's implementation of the waste package general corrosion model abstraction in the TSPA code is reasonable because it would not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Summary of Conclusions for the General Corrosion of the Waste Package Outer Barrier

The NRC staff reviewed the DOE model abstraction for general corrosion of the waste package outer barrier that was implemented in the TSPA code. DOE used reasonable experimental tests and other independent technical literature to provide adequate support for the model abstraction. Therefore, DOE reasonably accounted for general corrosion of the waste package outer barrier in the TSPA code.

2.2.1.3.1.3.2.2 Localized Corrosion of the Waste Package Outer Barrier

Localized corrosion is a process where corrosion occurs at discrete sites, in contrast to general corrosion, which uniformly thins the entire surface of a material. Localized corrosion usually occurs in metals and alloys, such as Alloy 22, whose corrosion resistance is attributed to the presence of a passive oxide film. Localized corrosion can initiate if the passive film is removed or damaged. When localized corrosion does occur, it tends to cause degradation much faster than general corrosion. In SAR Section 2.3.6.4, DOE considered that localized corrosion could lead to the release of radionuclides from the waste package if the waste package wall is breached.

DOE determined that localized corrosion requires the presence of a liquid water film on the waste package surface, which may come from dripping seepage water or salt deliquescence in dust particles, as outlined in SNL Section 6.3.5.2 (2008ag). DOE's evaluation of salt deliquescence indicated that brines produced from dust deposits will not lead to localized corrosion [features, events, and processes (FEP) 2.1.09.28.0A; SNL (2008ac)]. Consequently, DOE excluded localized corrosion caused by deliquescence from the performance assessment (SAR Table 2.2-5) and concluded that seepage water must contact the waste package for localized corrosion to occur.

In the TSPA code, DOE calculated that in-drift conditions (i.e., temperature, pH, and concentration of ionic species in seepage water) may support localized corrosion of the waste package for approximately 12,000 years after repository closure, as described in DOE Enclosure 1 (2009dg). The TSPA code, however, also calculates that few drip shields fail within 12,000 years after repository closure. Therefore, the probability of waste package breach by localized corrosion is low in the DOE model. Following 12,000 years after repository closure, DOE calculated that there is a low probability for conditions in the drift to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package.

In TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6, respectively, the NRC staff notes that chemical and mechanical degradation of the drip shield are very unlikely to cause failure of the drip shield plates and allow seepage water to contact the waste package within 12,000 years of repository closure. Therefore, there is a low probability of waste package breaches by localized corrosion within 12,000 years of repository closure. Nevertheless, there are uncertainties related to the barrier capability of the drip shield beyond 12,000 years after repository closure, and seepage water may contact the waste package during this period. As such, this section provides the NRC staff's review of the DOE model abstractions for initiation and propagation of localized corrosion, focusing on localized corrosion behavior beyond 12,000 years after repository closure.

Waste Package Localized Corrosion Conceptual Models

DOE implemented models for both initiation and propagation of localized corrosion of the waste package outer barrier in the TSPA code. The following addresses the NRC staff's review of these models.

Corrosion Initiation Models

DOE considered two potential mechanisms by which localized corrosion could be initiated on the waste package outer barrier under seepage conditions. In SAR Section 2.3.6.4, DOE described the first mechanism, which is related to the waste package open-circuit corrosion potential, or corrosion potential. The DOE model initiates localized corrosion if the corrosion potential for the waste package is greater than or equal to a critical potential. DOE defined critical potential as the potential above which a passive film will not spontaneously reform if damaged (SAR Section 2.3.6.4.1). While localized corrosion typically encompasses both pitting and crevice corrosion, DOE treated all waste package localized corrosion as crevice corrosion because crevice corrosion initiates in less aggressive thermal and chemical conditions than pitting corrosion, as described in SNL Section 6.4.4 (2007al). As such, DOE assumed that the critical potential for localized corrosion is equivalent to the crevice repassivation potential (SAR Section 2.3.6 4.1).

The second initiation mechanism, referred to as salt separation, was described in SAR Section 2.3.5.5.4.2.1. During salt separation, the relative humidity at the waste package surface drops below a salt precipitation threshold while seepage is occurring, causing chloride salts to precipitate out of solution. Nitrate that is still in solution moves away by advection. A chemically aggressive chloride-rich, nitrate-depleted brine forms when the relative humidity increases above this threshold value. DOE did not, however, model localized corrosion by salt separation in the TSPA code, as summarized in SAR Section 2.4. Consequently, the NRC staff submitted a request for additional information that DOE provide technical details evaluating the significance of salt separation effects on the performance assessment of waste packages in the proposed repository environment. DOE Enclosure 1 (2009dg) provided additional information indicating that the salt separation aspects of localized corrosion initiation were not implemented. DOE further stated that the information and analysis provided in DOE Enclosure 1 (2009dg) (i.e., that the salt separation aspects of localized corrosion initiation were not implemented) will be included in a future SAR update.

Summary of NRC Staff's Review for Corrosion Initiation Models

The NRC staff reviewed the DOE conceptual models for localized corrosion initiation on the waste package outer barrier, as described in SAR Sections 2.3.6.4 and 2.3.5.5. The NRC staff notes that DOE's model assumption that localized corrosion will initiate when the corrosion potential for the waste package is greater than or equal to a critical potential (i.e., the crevice repassivation potential) is consistent with the understanding of corrosion processes in the technical literature (Evans, et al., 2005aa,ab; Dunn, et al., 2000aa). Moreover, this initiation model will not underestimate the probability of localized corrosion initiation, because the initiation of Alloy 22 crevice corrosion generally requires less aggressive conditions than the initiation of pitting corrosion (Rebak, 2005aa; Cragnolino, et al., 1999aa; Dunn, et al., 2000aa). For initiation by salt separation, the NRC staff notes that DOE's model of chloride salt precipitation from brines in low humidity conditions is consistent with the thermodynamic physics of salt solutions (Yang, et al., 2006aa).

The NRC staff also notes that crevice corrosion is typically associated with small volumes of stagnant solution in holes, under surface deposits, or underneath fasteners (Fontana and Greene, 1978aa). Moreover, experimental data indicate that some crevice couples that could form in the repository do not support localized corrosion even if they form a tight crevice (He, et al., 2007ab; Shan and Payer, 2007aa). Alloy 22-to-Alloy 22, Alloy 22-to-titanium, and Alloy 22-to-ceramic couples have low susceptibility to crevice corrosion in concentrated sodium chloride solutions, which may be representative of the nitrate-depleted brine that forms after salt separation. Therefore, the DOE model that localized corrosion could occur on any part of the waste package surface exposed to seepage water will not underestimate that fraction of the waste package surface that undergoes localized corrosion.

Corrosion Propagation Model

In DOE's model for localized corrosion propagation, corrosion propagates at a constant rate over time (SAR Section 2.3.6.4.3.2). DOE also considered an alternative conceptual model in which the corrosion rate decreased over time (SAR Section 2.3.6.4.3.2.2). The alternative model, based on pit growth, gives a corrosion propagation rate lower than that calculated using the primary model assumption of constant corrosion rate. Therefore, DOE implemented the model with constant corrosion rate in TSPA code because it calculated an earlier waste package breach time.

Summary of NRC Staff's Review for Corrosion Propagation Model

The NRC staff reviewed the DOE model for propagation of localized corrosion of the waste package outer barrier, as described in SAR Section 2.3.6.4.3.2. DOE presented experimental evidence for a decreasing localized corrosion rate over time in a range of metals and alloys (Hunkeler and Boehni, 1983aa; Marsh, et al., 1991aa; Mughabghab and Sullivan, 1989aa; Sharland, et al., 1994aa; Ishikawa, et al., 1994aa). Also, the NRC staff notes that localized corrosion kinetics in Alloy 22 is likely to be slower than these materials because of its persistent passive film. DOE's use of the assumption of a constant localized corrosion rate over time for the waste package outer barrier is reasonable because it will not underestimate the corrosion propagation rate.

Localized Corrosion Initiation Conditions

For localized corrosion initiation by corrosion potential and salt separation, respectively, DOE used empirical data to establish the conditions in which localized corrosion of the waste package outer barrier could initiate. The NRC staff reviewed DOE's localized corrosion initiation conditions.

Initiation by Critical Potential

In the TSPA code, DOE compared the waste package outer barrier corrosion potential to the repassivation potential to determine whether electrochemical conditions on the waste package would lead to passive film breakdown. DOE assumed that the corrosion potential and repassivation potential for the waste package outer barrier, respectively, depended on the environmental conditions in the drift, including temperature, pH, chloride concentration, and nitrate concentration. DOE derived equations to represent the potentials as functions of these parameters by performing tests in which it measured the potentials for Alloy 22 specimens while varying the environmental parameters (SAR Section 2.3.6.4.2).

DOE established the dependence between the corrosion potential of Alloy 22 and the environmental parameters using data from 5-year tests (SAR Section 2.3.6.4.2.1). For a range of temperatures, DOE measured the corrosion potentials for Alloy 22 creviced samples with various metallurgical conditions (i.e., as welded, mill annealed, stress relieved) exposed to a range of simulated brine solutions, including simulated dilute water , simulated acidified water, and simulated concentrated water, the compositions of which were given in SAR Table 2.3.6-1. The corrosion potential data used in DOE's model development were shown in SAR Table 2.3.6-6. Using regression analyses, DOE applied SAR Eq. 2.3.6-7 to the data shown in SAR Table 2.3.6-6, representing the corrosion potential as a function of temperature, pH, and nitrate and chloride ion concentrations.

DOE established the dependence between the repassivation potential of Alloy 22 and the environmental parameters using data from cyclic potentiodynamic polarization tests (SAR Sections 2.3.6.4.2.2). The tests were performed using the methodology of ASTM G–61-86 (ASTM International, 2003aa). In DOE Enclosure 9 (2009cl), DOE stated that while repassivation potentials can be measured by methods other than cyclic potentiodynamic polarization, the differences in measured potentials tended to be small and the cyclic potentiodynamic polarization method generally predicted greater corrosion susceptibility in aggressive brines. Similar to DOE's tests for corrosion potential, DOE used Alloy 22 specimens with different metallurgical conditions (mill annealed and as welded) in a range of simulated brine solutions. DOE used ceramic wrapped with polytetrafluoroethylene tape to form a crevice with Alloy 22 in the cyclic potentiodynamic polarization tests. The crevice repassivation potential data used in DOE's model development were shown in SAR Table 2.3.6-7. Sample specimens that did not show evidence of localized corrosion attack were not used to develop the model. For experiments showing localized corrosion, DOE used regression analyses to fit SAR Eq. 2.3.6-6 to the data shown in SAR Table 2.3.6-7, representing the crevice repassivation potential as a function of temperature and nitrate and chloride ion concentrations. In the TSPA code, DOE accounted for fitting uncertainty in SAR Eq.s 2.3.6-6 and 2.3.6-7 by varying the values of the fitting parameters of the respective equations according to a Monte Carlo algorithm and by using data in the regression analysis from multiple samples for a given environmental condition, as detailed in SNL Section 6.3.5.2.2 (2009ag).

DOE compared the model calculations of corrosion potential and repassivation potential, respectively, to establish the environmental conditions that would support initiating localized corrosion on the waste package outer barrier (SAR Section 2.3.6.4.3.1.2). In the DOE model, the probability for initiation of localized corrosion at temperatures less than 90 °C [194 °F] generally increases with decreasing pH and decreasing nitrate-to-chloride ion ratio. DOE compared the localized corrosion initiation conditions to experimental observations of localized corrosion initiation on Alloy 22 specimens. DOE's model calculated that localized corrosion may initiate on Alloy 22 exposed to simulated acidified water at 90 °C [194 °F], whereas in experimental tests, localized corrosion was not observed on Alloy 22 specimens exposed to this solution for 5 years (SAR Table 2.3.6-12). DOE presented a number of additional corrosion test solutions for which its model calculated that localized corrosion may initiate, yet for which localized corrosion was not observed on Alloy 22 specimens (SAR Table 2.3.6-13). DOE concluded that its model overestimates the probability of localized corrosion initiation.

Summary of NRC Staff's Review for Initiation by Critical Potential

The NRC staff reviewed DOE's experimental methodology to measure the corrosion potential and the crevice repassivation potential of Alloy 22. The NRC staff notes that DOE's tests were consistent with the general practice for measuring the corrosion potential and repassivation potential of metallic materials. In particular, the cyclic potentiodynamic polarization method is a reasonable way to measure the repassivation potential because it predicts higher corrosion susceptibility than other measures of this parameter, as confirmed by independent analyses by the NRC staff (He, et al., 2009aa). Also, the NRC independent analyses (He, et al., 2007aa) and Shan and Payer (2007aa) suggest that ceramic wrapped with polytetrafluoroethylene tape, which DOE used to form a crevice with Alloy 22 in the cyclic potentiodynamic polarization tests, is a material combination that is more favorable for localized corrosion than many combinations that would be expected in the repository, including Alloy 22-to-Alloy 22, Alloy 22-to-titanium, and Alloy 22-to-ceramic couples.

The NRC staff reviewed DOE's calculated values for the corrosion potential and repassivation potential, respectively, to confirm the environmental conditions (i.e., temperature, pH, concentration of ionic species in seepage water) under which the DOE model calculates that localized corrosion of the waste package will initiate. The NRC staff examined the model functional dependencies and confirmed that in the DOE model, the probability for localized corrosion initiation at a given temperature generally increases with decreasing pH and decreasing nitrate-to-chloride ion ratio. Therefore, the NRC staff notes that the DOE model may calculate that localized corrosion initiates in acidic solutions such as simulated acidified water or in solutions with very low nitrate-to-chloride ratios (SAR Tables 2.3.6-12 and 2.3.6-13). Nevertheless, Alloy 22 specimens did not show evidence of localized corrosion initiation during 5 years' immersion in a range of corrosion test solutions, including simulated acidified water, at temperatures up to 90 °C [194 °F] (SAR Table 2.3.6-12) and that for shorter tests localized corrosion was only observed in concentrated chloride brines (SAR Table 2.3.6-13). On the basis of these data, Alloy 22 is resistant to localized corrosion in brine solutions that are more chemically aggressive than the waters expected to occur within the repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5 and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. As such, the DOE model is reasonable because it calculates a higher probability of localized corrosion initiation than is expected based on experimental data.

Initiation by Salt Separation

DOE used thermodynamic analyses to calculate the threshold relative humidity below which chloride-bearing salts could precipitate out of seepage water (SNL, 2007ak). In these analyses, DOE considered that the threshold relative humidity depends upon the group water type (i.e., 1–4 as defined in SAR Section 2.3.5), quantity of alkali feldspar to be tritiated into the seepage waters [i.e., the water–rock interaction parameter (WRIP)], the partial pressure of CO_2 in the drift, and the waste package temperature. In SAR Figure 2.3.5-55, the thermodynamic analyses showed that the chloride-to-nitrate ratio in a range of conditions is nearly constant until the activity of water (i.e., relative humidity) drops below a value in the range of approximately 77 to 65 percent. In the TSPA model, DOE concluded that this range represents the threshold relative humidity below which localized corrosion initiation by salt separation can occur (SAR Section 2.3.5.5).

Summary of NRC Staff's Review for Initiation by Salt Separation

The NRC staff reviewed the DOE approach for establishing the relative humidity threshold for the initiation of localized corrosion by salt separation, as DOE described in SAR Section 2.3.5.5. The methodology DOE used to determine the threshold relative humidity is reasonable because it is based on well-established concepts regarding the thermodynamic stability of aqueous solutions. In particular, the threshold values DOE calculated are consistent with calculations for pure sodium chloride by Greenspan (1977aa), which are incorporated into ASTM E104-02 (ASTM International, 2007aa). Also, the NRC staff performed independent tests and observed that even in pure 5 M sodium chloride solution, localized corrosion was not initiated on Alloy 22 specimens in open circuit conditions without the addition of copper chloride as an oxidant (He and Dunn, 2005aa). The DOE model assumes that salt separation can cause the corrosion potential to exceed the repassivation potential. In experimental tests, however, DOE did not observe localized corrosion in many cases where the corrosion potential was greater than the repassivation potential, including 5-year tests in simulated acidified water (SAR Tables 2.3.6-12 and 2.3.6-13). On the basis of this information, the threshold relative humidity that DOE calculated is reasonable because it will not underestimate the probability of localized corrosion initiation.

Localized Corrosion Propagation Rate

In the Total System Performance Assessment (TSPA) code, DOE sampled the propagation rate for localized corrosion on the waste package outer barrier using a log-uniform distribution in the range of 12.7 to 1,270 μm/yr [5×10^{-4} to 5×10^{-2} in/yr] with a median value of 127 μm/yr [5×10^{-3} in/yr] (SAR Section 2.3.6.4.3.2). This range was based on corrosion testing of Alloy 22 in aggressive environments, including 10 percent $FeCl_3$ (Haynes International, 1997aa) and concentrated HCl (Haynes International, 1997ab). DOE compared the corrosion rate distribution sampled in the TSPA code to independently measured corrosion rates for similar, but less corrosion-resistant alloys, including Alloy C-276 and Alloy C-4 (SAR Section 2.3.6.4.4.2.2). DOE concluded that the measured corrosion rates fall within the bounds of the distribution sampled in TSPA code.

Summary of NRC Staff's Review for Localized Corrosion Propagation Rate

The NRC staff reviewed the DOE approach, described in SAR Section 2.3.6.4, that established the distribution sampled in the TSPA code for the propagation rate of localized corrosion on the waste package outer barrier. The NRC staff determined that the distribution sampled in the TSPA code is based on corrosion data from studies that considered thermal and chemical conditions that were more aggressive than those expected in the repository. Also, DOE's calculated range for the corrosion rate is consistent with the NRC staff independent measurements of the Alloy 22 localized corrosion rate (He and Dunn, 2005aa). On the basis of this information, the distribution from which DOE sampled the waste package outer barrier localized corrosion propagation rate in the TSPA code is reasonable because it will not underestimate the propagation rate. Localized corrosion at the rates DOE calculated would penetrate the 25-mm [0.98-in]-thick waste package outer barrier in approximately 20 to 2,000 years, which is short relative to a 10,000-year period. Therefore, the NRC staff notes that uncertainties in the localized corrosion rate would not significantly affect the timing or magnitude of radionuclide release.

Effects of Microorganisms on Localized Corrosion

As DOE addressed in SAR Section 2.3.6.3.2.3, microorganisms in the repository may affect the corrosion processes on the waste package outer barrier such that the type and extent of corrosion in the presence of the microorganisms may be different from the corrosion in the absence of the microorganisms. Though DOE incorporated microbially influenced corrosion effects into its model abstraction for general corrosion of the waste package outer barrier (SAR Section 2.3.6.3.2.3), experimental observations indicate that microbially influenced corrosion may also affect the localized corrosion behavior. In particular, a DOE study showed that small pits, or micropores, were observed on the surface of Alloy 22 corrosion specimens exposed in a borosilicate glass vessel with unsterilized Yucca Mountain tuff rock, whereas no such micropores were observed in sterilized conditions (Martin, et al., 2004aa). In DOE Enclosure 10 (2009cl), DOE stated that the micropores Martin, et al. (2004aa) observed started to form during the first 17 months' exposure, after which the size of the micropores was less than 1 μm [0.039 mil] in diameter. DOE further stated that the same specimens were observed after an additional 40 months' exposure, after which there were more pores, but no significant increase in pore size compared to that measured after 17 months. DOE determined that, if the micropores were initiated pits, they quickly repassivated before they could propagate. Therefore, DOE concluded that it was appropriate to incorporate microbially influenced corrosion into its model abstraction for general corrosion of the waste package outer barrier, to be consistent with DOE observations in SNL Section 6.4.5 (2007al) that microbially influenced corrosion may enhance corrosion on the entire material surface.

Summary of NRC Staff's Review for Effects of Microorganisms on Localized Corrosion

The NRC staff reviewed the information in DOE Enclosure 10 (2009cl). The DOE analysis that the pores may represent initiated pits that quickly repassivated is reasonable. Because the pores did not grow after repassivation, the NRC staff notes that the micropores are not indicative of localized corrosion that could significantly affect the timing or magnitude of radionuclide release from the waste package. The NRC staff also identified an independent report which confirmed that Alloy 22 is highly resistant to localized corrosion in microorganism-rich environments, including seawater, which also has low nitrate content (Aylor, et al., 1999aa). On the basis of this information, DOE's exclusion of microbially influenced corrosion effects in the model abstraction for localized corrosion of the waste package outer barrier is reasonable.

Abstraction and Integration

DOE did not directly include and calculate the effects of localized corrosion of the waste package outer barrier in the TSPA code. Rather, DOE performed a Localized Corrosion Initiation Uncertainty Analysis, as described in SNL Appendix O (2008ag) and DOE Enclosure 1 (2009dg), that calculated the fraction of waste packages in the repository that are susceptible to localized corrosion as a function of drip shield breach time (i.e., the time at which seepage water could contact the waste package). The Localized Corrosion Initiation Uncertainty Analysis implements the Localized Corrosion Initiation Submodel, as detailed in SNL Section 6.3.5.2.3 (2008ag), which determines whether the environmental conditions in the drift will initiate localized corrosion, and gives this input to the TSPA code. The Localized Corrosion Initiation Submodel is similar to the TSPA code, but incorporates only those submodels that are needed to calculate localized corrosion initiation conditions. In particular, the Localized Corrosion Initiation Submodel uses information primarily from the

- Engineered Barrier System Thermohydrologic Environment Submodel to determine the temperature and relative humidity history at the waste package

- Drift Seepage Submodel to determine whether seepage occurs at a repository location

- Engineered Barrier System Chemical Environment Submodel to determine the chemical composition of seepage water

- Seismic Ground Motion Damage Submodel to determine the time of drip shield plate failure due to seismic damage

- Waste package and Drip Shield Degradation Submodel to account for corrosion in determining drip shield plate and waste package failure times

In the Localized Corrosion Initiation Submodel, the repository was discretized into 3,264 subdomains of equal area, at the center of which were 6 commercial spent nuclear fuel and 2 codisposal waste packages. The subdomains were distributed through the five percolation subregions. In the Localized Corrosion Initiation Submodel, localized corrosion could initiate because of the waste package corrosion potential or by salt separation. For each subregion, at every timestep in a realization, the Localized Corrosion Initiation Submodel compares the corrosion potential, as calculated by SAR Eq. 2.3.6-7, to the crevice repassivation potential, as calculated by SAR Eq. 2.3.6-6. If the corrosion potential was greater than or equal to the crevice repassivation potential in seepage conditions, the Localized Corrosion Initiation Submodel assumed that localized corrosion initiated at that subregion. Similarly, if the relative humidity at the waste package surface fell below the salt separation threshold humidity in seepage conditions, the Localized Corrosion Initiation Submodel assumed that localized corrosion initiated at that subregion.

The Localized Corrosion Initiation Submodel calculated that, if the drip shield were breached at the time of the repository closure (i.e., no drip shield present), there is approximately 34 percent probability that localized corrosion will initiate on a given waste package surface (24 percent probability contribution by salt separation and 10 percent probability contribution by corrosion potential), as shown in DOE Enclosure 1, Figure 1 (2009dg). DOE calculated that as the time to drip shield breach increases, the probability of localized corrosion initiation decreases and is negligible if the drip shield fails beyond approximately 12,000 years after repository closure. In particular, DOE calculated that localized corrosion will not initiate by salt separation if drip shield breach occurs after approximately 1,000 years from the time of repository closure, because the relative humidity will remain above the threshold value. DOE also stated that changes in the repository environmental and chemical conditions (e.g., decreasing temperature) make initiation by corrosion potential less probable as the time to drip shield breach increases in its model.

Given the results of the Localized Corrosion Initiation Uncertainty Analysis, DOE concluded that localized corrosion of the waste package outer barrier could affect the timing and magnitude of the release of radionuclides from the waste package only if the overlying drip shield plate was breached within approximately 12,000 years after repository closure. In regard to drip shield early failure, DOE assumed that localized corrosion under seepage conditions occurs on the waste packages located beneath the failed drip shield. Other than drip shield early failure, DOE modeled that no drip shield failure would occur within the approximately 12,000 years. Therefore, as shown in DOE Enclosure 1 (2009dg), DOE concluded that localized corrosion of the waste package outer barrier will have no significant effect upon either the timing or magnitude of radionuclide release calculated in the TSPA modeling cases.

Summary of NRC Staff’s Review for Abstraction and Integration for Localized Corrosion

The NRC staff reviewed the implementation and integration of the model abstraction for localized corrosion of the waste package outer barrier in the postclosure performance assessment. DOE has provided information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that the model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the staff notes that, with respect to the parameters used in the model abstraction (i.e., corrosion potential, corrosion repassivation potential, relative humidity, and pit growth rate), DOE adequately justified the ranges of these parameters and accounted for uncertainty in the model abstraction.

DOE's analysis is reasonable that the consequence of drip shield breach within 12,000 years on the overall dose is negligible because the analysis does not underestimate the onset of seismic-induced breach. The NRC staff also notes that the probability of waste package localized corrosion initiation beyond 12,000 years after repository closure is low, even if drip shield failure allows seepage water to contact the waste package. In the Localized Corrosion Initiation Uncertainty Analysis, the probability for localized corrosion initiation decreases with increasing pH and increasing nitrate-to-chloride ion ratio. DOE showed that the pH and nitrate-to-chloride ion ratio of in-drift waters will generally be too high to initiate localized corrosion beyond 12,000 years after repository closure. The DOE model also shows that localized corrosion may initiate in low pH solutions or solutions with low nitrate-to-chloride ratio. DOE's experimental data, however, showed that localized corrosion does not initiate at 90 °C [194 °F], even in corrosion test solutions with lower pH or lower nitrate-to-chloride ratio than waters expected to be present in drifts at such temperatures, including starting seepage waters used by DOE (SAR Section 2.3.5.5) and waters considered in the NRC staff's independent analysis of in-drift water evolution (TER Section 2.2.1.3.3.3.2). As such, the NRC staff notes that waste package breach by localized corrosion is unlikely beyond 12,000 years after repository closure, even if drip shield failure allows seepage water to contact the waste package. The implementation of the TSPA model abstraction for localized corrosion of the waste package outer barrier is reasonable because it would not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Summary of Conclusions for Localized Corrosion of the Waste Package Outer Barrier

The NRC staff reviewed the DOE models for localized corrosion of the waste package outer barrier that were implemented in the TSPA code. DOE used appropriate experimental tests and applicable technical literature to provide support for the localized corrosion initiation and propagation models. For the first 12,000 years after repository closure, the NRC staff noted that waste package breach by localized corrosion is unlikely because the intact drip shields will prevent seepage waters from contacting the waste package (TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6). For the time period beyond 12,000 years after repository closure, the DOE models show a low probability for localized corrosion initiation because the repository environment (i.e., temperature, pH, and chemical composition of seepage waters) will not support the initiation of localized corrosion. Therefore, DOE's analytic models for localized corrosion of the waste package outer barrier in the postclosure performance assessment analyses are reasonable.

2.2.1.3.1.3.2.3 Stress Corrosion Cracking of the Waste Package Outer Barrier

Stress corrosion cracking generally refers to a process whereby cracks form in metals or alloys in a corrosive environment and under sustained tensile stresses. DOE presented data indicating that Alloy 22 is highly resistant to stress corrosion cracking in the environmental conditions (e.g., temperature, pH, and chemical constituents of seepage water brines) that are expected to occur in the repository, as detailed in SAR Section 2.3.6.5.2 and SNL Section 6.2 (2007bb). Because of uncertainty regarding the long-term environmental conditions in the repository, however, the DOE model in the TSPA code assumes that the repository environment supports stress corrosion cracking, such that sufficient residual tensile stress was the only criterion for stress corrosion cracking occurrence (SAR Section 2.3.6.5.1). In SAR Section 2.3.6.5, DOE evaluated stress corrosion cracking caused by residual stresses from waste package fabrication. In SAR Section 2.3.4.5, DOE also evaluated stress corrosion cracking caused by the residual stresses resulting from impacts to the waste package during seismic ground motions. This section of the TER includes NRC staff's review of the DOE model abstractions for stress corrosion cracking of the waste package outer barrier.

Conceptual Models

The DOE models for stress corrosion cracking of the waste package outer barrier treat crack initiation (i.e., the formation of cracks on the waste package surface) and crack propagation (i.e., growth of cracks from the surface through the outer barrier) as distinct phenomena. In the TSPA code, DOE assumed that cracks initiated on areas of the waste package surface where the magnitude of the sustained tensile stress was greater than a threshold value, which DOE referred to as the residual stress threshold (RST) (SAR Section 2.3.6.5.2.1). Cracks initiated by this sustained stress were referred to as incipient cracks. In the DOE model, the residual stresses for crack initiation could result from fabrication processes, such as welding, or from impacts to the waste package during a seismic ground motion event. DOE stated that the concept of a threshold stress that must be exceeded for the onset of stress corrosion cracking is widely accepted in the technical literature (e.g., ASM International, 1987aa). In the TSPA code, DOE also assumed that waste package weld flaws (e.g., voids and slag inclusions) were initiated cracks, regardless of the magnitude of the residual stress (SAR Section 2.3.6.5.3.1).

DOE used different conceptual models for the propagation of cracks initiated by fabrication stresses and weld flaws and those initiated by seismically induced stresses, respectively. With regard to cracks initiated by seismically induced stresses, DOE did not explicitly model crack propagation. Rather, DOE assumed that cracks instantaneously propagated through the wall at the time of initiation SAR Section 2.3.4.5.1.2.1. This assumption minimizes the time for through-wall propagation of cracks. With regard to cracks initiated by fabrication stresses and weld flaws, DOE assumed that the stress intensity factor at the tip of the initiated crack must be greater than a threshold value for the crack to propagate (SAR Section 2.3.6.5.3.2). DOE stated that the concept of a threshold stress intensity factor is consistent with the general understanding of crack fracture mechanics (Jones and Ricker, 1987aa; Sprowls, 1987aa).

To calculate the rate of growth for cracks with a stress intensity factor greater than the threshold value, DOE used the slip-dissolution film-rupture (SDFR) model, as discussed in SAR Section 2.3.6.5.3.2 and SNL Section 6.4.2 (2007bb). In the slip-dissolution film-rupture model, crack growth is related to the rupture and subsequent reformation of the passive metal oxide film at the crack tip. DOE stated that several studies (Ford and Andresen, 1988aa; Andresen, 1991aa; Andresen and Ford, 1994aa) used the slip-dissolution film-rupture model to accurately calculate crack growth rates in stainless steel and nickel-based alloys similar to Alloy 22 (e.g., Alloys 182 and 600). In SAR Section 2.3.6.5.3.3, DOE described an alternative conceptual model for the crack growth rate: the coupled environmental fracture model. The coupled environmental fracture model is based on conservation of electrons involved in the corrosion process (Macdonald and Urquidi-Macdonald, 1991aa; Macdonald, et al., 1994aa). It incorporates the effects of oxygen concentration, flow rate, and the conductivity of the external environment and accounts for the effect of stress on crack growth. DOE did not use the coupled environmental fracture model in the TSPA calculation, because it calculated a slower crack growth rate than the slip-dissolution film-rupture model.

Summary of NRC Staff's Review for Conceptual Models for Stress Corrosion Cracking

The NRC staff reviewed DOE's models for stress corrosion cracking initiation. DOE's use of a threshold stress for the onset of stress corrosion cracking in the waste package outer barrier is reasonable because this concept is consistent with reports of stress corrosion cracking behavior in a range of passive alloys similar to Alloy 22 (e.g., ASM International, 1987aa). Phenomenologically, the NRC staff notes that the initiation of stress corrosion cracking in Alloy 22 is similar to initiation in other passive alloys involving repetitive rupture and regeneration of a passive film. The NRC staff also notes that, because weld flaws may be present in the waste package outer barrier at the time of emplacement, it is reasonable for DOE to model the flaws as initiated cracks. Therefore, the DOE conceptual models for crack initiation are reasonable.

The NRC staff also reviewed DOE's models for stress corrosion cracking propagation. Regarding the propagation of cracks initiated by seismically induced stresses, the NRC staff notes that the DOE model does not take credit for the possibility that initiated cracks could arrest before propagating through the barrier or the time it would take for cracks to pass through the barrier. Therefore, the DOE model for propagation of seismically induced cracks is reasonable because it will underestimate the time it takes for cracks to breach the waste package outer barrier. Regarding the propagation of weld flaws and cracks initiated by fabrication stresses, DOE's use of a threshold stress intensity factor is consistent with the technical understanding of crack fracture mechanics. Further, DOE has validated the slip-dissolution film-rupture model predictions with experimental results (SAR Figure 2.3.6-29). Therefore, the use of slip-dissolution film-rupture model for calculating the crack growth rates is reasonable. The staff also notes that the slip-dissolution film-rupture model calculates higher crack growth rates than DOE measured using the reversing direct current measurement technique on compact-tension-type Alloy 22 fracture mechanics specimens (SAR Figure 2.3.6-34), or greater than was calculated by the alternative coupled environmental fracture model (Ford and Andresen, 1988aa). As such, the DOE models for propagation of weld flaws and cracks initiated by fabrication stresses are reasonable because they will not overestimate the time it takes for cracks to breach the waste package outer barrier.

Stresses for Crack Initiation and Propagation

The NRC staff reviewed the DOE approaches to establish residual stress threshold values for crack initiation and the threshold stress intensity factor for the propagation of weld flaws and cracks initiated by fabrication stresses.

Residual Stress Threshold

DOE performed laboratory tests to establish the value of the residual stress threshold for Alloy 22. DOE performed constant-load crack initiation tests (SAR Section 2.3.6.5.2.1.1), slow strain rate tests (SAR Section 2.3.6.5.2.1.2), and U-bend stress corrosion cracking initiation tests (SAR Section 2.3.6.5.2.1.3). These tests were performed for up to 5 years for Alloy 22 specimens with metallurgical conditions representative of waste package metallurgical conditions in the repository (i.e., welded, thermally aged, cold worked). The tests were performed in the temperature range of 25 to 165 °C [77 to 329 °F] in different brines, including basic simulated water, simulated dilute water, simulated concentrated water, and simulated acidified water, the compositions of which are shown in SAR Table 2.3.6-1.

For the constant-load crack initiation tests (SAR Section 2.3.6.5.2.1.1), DOE exposed Alloy 22 specimens to basic simulated water (pH of 10.3) at 105 °C [221 °F] for up to 28,000 hours (approximately 3 years). The test specimens were subjected to tensile stress up to 2.1 times the at-temperature yield strength for as-received materials and 2.0 times the yield strength of the welded materials, which corresponds to approximately 95 percent of the ultimate tensile strength of Alloy 22 in the respective material conditions. DOE reported that no sample ruptured during the test, as shown in SAR Figure 2.3.6-28. For the slow strain rate testing (SAR Section 2.3.6.5.2.1.2), Alloy 22 specimens were exposed to simulated acidified water, basic simulated water, simulated concentrated water, and calcium-chloride-type brines over a range of temperatures, with and without applied potential (SAR Table 2.3.6-14). DOE stated that it did not observe stress corrosion cracking in most experimental conditions, though stress corrosion cracking was observed in simulated concentrated water with large applied anodic potentials. DOE concluded in SNL Section 6.2.1.3 (2007bb), however, that such potentials are not representative of repository conditions. For the U-bend stress corrosion cracking initiation tests (SAR Section 2.3.6.5.2.1.3), Alloy 22 specimens were tested in simulated dilute water, simulated concentrated water, and simulated acidified water for 5 years with no evidence of stress corrosion cracking initiation.

Even though stress corrosion cracking of Alloy 22 was not observed in the experimental testing, DOE concluded that cracks may initiate at lower stresses on the repository time scale, yet would not be observed in short-term laboratory tests, as described in SNL Section 6.2.2 (2007bb). Therefore, DOE stated that there existed some uncertainty associated with the value of the residual stress threshold. Thus, to establish the residual stress threshold value for the TSPA code, DOE applied a safety factor of two to the maximum stress that Alloy 22 specimens withstood with no evidence of stress corrosion cracking initiation. As outlined in SNL Section 6.2.2 (2007bb), DOE determined that this maximum stress was 210 percent of the Alloy 22 at-temperature yield strength, as measured during constant-load crack initiation testing in basic simulated water. This approach established the upper bound for the residual stress threshold to be 105 percent of the Alloy 22 at-temperature yield strength. DOE stated that use of a safety factor of two is consistent with general engineering practice and has been used to establish the allowable long-term fatigue stress on engineering components (American Society of Mechanical Engineers, 1969aa). To further account for uncertainty in the residual stress threshold, DOE established 90 percent of the Alloy 22 at-temperature yield stress as a lower bound. Thus, in the Total System Performance Assessment (TSPA) code, DOE sampled the residual stress threshold from a uniform distribution between 90 and 105 percent of the Alloy 22 at-temperature yield stress, as shown in SNL Table 6-3 (2007bb).

In response to the NRC staff's request for additional information that DOE assess the observed case of stress corrosion cracking initiation in simulated concentrated water, DOE provided new data in DOE Enclosure 1 (2009cj) from U-bend testing of as-welded and mill-annealed Alloy 22 specimens at 165 °C [329 °F] in aerated simulated concentrated water (Andresen and Kim, 2009aa). After 32,000 hours (approximately 3.5 years), no stress corrosion cracking was observed for stresses estimated to be at or slightly above the at-temperature yield strength of Alloy 22. DOE cited an additional study in which low strain rate crack initiation tests were performed on Alloy 22 specimens in simulated concentrated water at 86 and 89 °C [186.8 and 192.2 °F] (Fix, et al., 2003aa). DOE reported that crack initiation did not occur until the tensile stress exceeded a value of 605 MPa [87.7 ksi], which is approximately 160 percent of the room temperature yield strength of Alloy 22. On the basis of this information, DOE concluded that the range of the residual stress threshold, given in SAR Section 2.3.6.5, was adequate.

DOE evaluated the expected fabrication-induced residual stresses during the postclosure period to determine whether such stresses could exceed the residual stress threshold. DOE stated that it plans to do a stress-relief heat treatment to mitigate the stresses in the waste package outer barrier after fabrication (SAR Section 1.5.2.7.1) following the standards specified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC-4600 (American Society of Mechanical Engineers, 2001aa). DOE concluded that fabrication-induced residual stresses will not exceed the residual stress threshold for portions of the waste package that are heat treated. In the DOE fabrication process, however, the heat treatment takes place before the waste is placed in the waste package and the outer lid is welded onto the shell. Welding the outer lid onto the waste package shell may induce residual stresses in the region of the closure weld that the heat treatment process cannot mitigate, as described in SNL Section 6.5.3.1 (2007bb). Therefore, DOE concluded that the region of the waste package closure weld is the only part of the waste package outer barrier where fabrication-induced stresses could cause stress corrosion cracking. DOE plans to implement a process called low plasticity burnishing (a process whereby the material surface is plastically deformed to create a layer with compressive residual stress) to delay the initiation of incipient stress corrosion cracking (SAR Section 1.5.2.7.2.2). The current waste package design requires a compressive residual stress to a depth of at least 3 mm [0.12 in] below the weld surface (SAR Table 1.9-9, Design Control Parameter 03-17). DOE concluded that the initiation of stress corrosion cracking would be delayed by the time it would take for general corrosion to corrode through at least the 3-mm [0.12-in] burnished layer.

DOE performed finite element analyses to calculate the residual stress profile of the weld, as detailed in SAR Section 2.3.6.5.2.3 and SNL Section 6.5.3.3 (2007bb), when the closure weld was plasticity burnished resulting in compressive stresses to a depth of 3 mm [0.12 in] below the weld surface. The analysis simulated multipass welds, with the residual stress represented as a function of welding parameters, thermal transients, temperature-dependent material properties, and elastic–plastic stress reversals. The DOE analyses indicated that the residual stress decays rapidly with increasing radial distance from the weld line and is negligible at a distance from the weld line approximately equal to the thickness of the waste package wall, as shown in SNL Figures 6-19 through 6-22 (2007bb). Given the rapid decay in weld-induced stress with increasing distance from the weld line, DOE assumed that initiation of stress corrosion cracking by fabrication-induced stress could only occur in patches representing the waste package closure weld in the TSPA model. These patches represent approximately 2.67 percent and 2.95 percent of the total surface area for the commercial spent nuclear fuel and the codisposal waste packages, respectively, as outlined in SNL Section 6.3.5.1.2 (2008ag).

In the region of the closure weld, DOE calculated that radial stresses do not exceed the residual stress threshold through the entire thickness of the weld, but that hoop stresses can exceed the residual stress threshold at a depth of approximately 5 to 7.5 mm [0.20 to 0.30 in] below the weld surface, as described in SNL Section 6.5.5.2.2 (2007bb). In the TSPA code, DOE represents the hoop stress as a function of depth from the weld surface in SNL Eq. 64 (2007bb). DOE also considered angular variability in the residual stress around the circumference of the waste package closure weld in SNL Eq. 6.3.5-6 (2008ag). On the basis of literature reports (e.g., Shack, et al., 1980aa), DOE calculated that the residual stress may have circumferential variation up to ± 2.5 ksi [± 17.24 MPa] from the mean stress, as detailed in SNL Section 6.5.6.1 (2007bb). SAR Figure 2.3.6-30 showed the angular variability in the residual hoop stress profile in the waste package closure weld. More generally, DOE identified literature reports (e.g., Mohr, 1996aa; Pasupathi, 2000aa) which indicated that welding and stress mitigation processes introduce uncertainty into the weld residual stress profile. The reports indicated that the uncertainty range for residual stress may be between 5 percent and 35 percent of the material yield strength. On the basis of the fabrication techniques and process controls planed for the waste package closure weld, DOE selected a three standard deviation uncertainty range of ±15 percent of the at-temperature yield strength of Alloy 22, as outlined in SNL Section 6.5.6.2 (2007bb). This is implemented in the TSPA model by applying a scaling factor to the residual stress. The scaling factor is sampled from a truncated (± 3 standard deviations) normal distribution where the mean is 0 and the standard deviation is 5 percent of the Alloy 22 at-temperature yield strength. SAR Figure 2.3.6-32 showed the uncertainty in the residual hoop stress profile in the waste package closure weld.

DOE compared the residual stress profile calculated by the finite element analysis to the residual stress experimentally measured by Woolf (2003aa) for plasticity-burnished Alloy 22 simulated closure welds, as described in SNL Section 6.5.6.5 (2007bb). As shown in SNL Figure 6-60 (2007bb), Woolf (2003aa) measured compressive residual stress to a depth of more than 7 mm [0.28 in] from the weld surface. DOE concluded that the calculated residual stress profile implemented in the TSPA code underestimates the extent of stress mitigation by plasticity burnishing.

Summary of NRC Staff's Review for Residual Stress Threshold

The NRC staff reviewed the DOE approach to establish the value of the residual stress threshold for the waste package outer barrier. The NRC staff notes that DOE used different types of stress corrosion cracking initiation tests, each of which is appropriate for measuring stress corrosion cracking susceptibility and is consistent with applicable standards, including ASTM G 30-94 (ASTM International, 1994aa); ASTM E 399-90, (ASTM International, 1991aa); ASTM G 129-00 (ASTM International, 2000ab); and ASTM G 49-85 (ASTM International, 2000aa). With regard to the material conditions for the stress corrosion cracking initiation tests, DOE tested Alloy 22 specimens with microstructures that are representative of those expected for the waste package on the basis of the fabrication procedures set forth in SAR Section 1.5.2.7. With regard to the test solutions for the corrosion tests, DOE tested Alloy 22 specimens in a range of solutions, including simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water   On the basis of information published by Chiang, et al. (2005aa, 20006aa) and Shukla, et al. (2006aa), the residual stress threshold for the waste package in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to measure the residual stress threshold in repository conditions. The NRC staff notes that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in the DOE near-field chemistry model described in SAR Section 2.3.5.5 and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in TER Section 2.2.1.3.3.3.2. As such, it is reasonable for DOE to measure the residual stress threshold on the basis of tests in these simulated brines.

The NRC staff reviewed DOE's use of the data from the stress corrosion cracking tests to establish the range for the residual stress threshold. The NRC staff notes that stress corrosion cracking initiation was observed only on Alloy 22 in simulated concentrated water under applied potential. This condition is not representative of repository conditions, because experimental studies showed that Alloy 22 specimens submerged in simulated concentrated water underwent stress corrosion cracking only when the applied potential was higher than the corrosion potentials (Fix, et al., 2003aa; Chiang, et al., 2005aa, 2006aa; Shukla, et al., 2006aa). Therefore, DOE did not observe stress corrosion crack initiation in any condition representative of the repository environmental conditions. As such, the NRC staff notes that reducing the stress that Alloy 22 withstood without evidence of stress corrosion cracking by a safety factor of 2 is reasonable to establish the upper bound for the residual stress threshold to be 105 percent of the at-temperature yield stress. Moreover, a residual stress threshold lower bound of 90 percent of the at-temperature yield stress appropriately quantifies the range of uncertainty associated with the value of this parameter. The distribution from which DOE sampled the residual stress threshold in the TSPA code is reasonable because it is unlikely to overestimate the stress at which stress corrosion cracking initiates in the waste package outer barrier.

The NRC staff reviewed the DOE analysis of fabrication-induced stresses in the waste package outer barrier. The NRC staff notes that a heat treatment process that follows the standards specified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC-4600 (American Society of Mechanical Engineers, 2001aa) is consistent with nuclear industry practice to relieve fabrication-induced stresses in components fabricated with materials similar to Alloy 22. Therefore, DOE consideration in the TSPA that stress corrosion cracking caused by fabrication-induced stresses could only occur in the region of the waste package closure weld, which is not heat treated, is reasonable.

The NRC staff also reviewed DOE's calculation of the closure weld residual stress profile, as described in SAR Section 2.3.6.5.2.3. DOE's finite element stress analyses were performed with a well-established methodology that is accepted in the technical literature (e.g., NRC, 1977aa; Rybicki and Stonsifer, 1979aa). Further, the NRC staff notes that the DOE-calculated residual stress profile is consistent with literature reports which show that residual stresses in highly controlled welds tend to persist for only a short distance from the weld line (ASM International, 1993aa). The NRC staff also notes that DOE provided information that shows that plasticity burnishing is an effective stress mitigation technique in engineered components (Prevey and Cammett, 2001aa) and that the stresses DOE calculated are lower than measured values (Woolf, 2003aa) for plasticity burnished Alloy 22 welds. Finally, the uncertainty and variability in the residual stress profile implemented in the TSPA code is reasonable because the model is consistent with weld stress analyses reported in the technical literature (Mohr, 1996aa; Pasupathi, 2000aa). Therefore, the residual stress profile for the waste package outer barrier closure weld used in the TSPA code is reasonable because it will not underestimate the residual stress.

Threshold Stress Intensive Fracture

DOE determined the numerical value of the stress intensity factor threshold using a crack blunting criterion, as detailed in SAR Section 2.3.6.5.1 and SNL Section 6.4.5 (2007bb). According to the crack blunting criterion, crack growth will arrest if the crack tip radius decreases, because the general corrosion rate at the sides of the crack is greater than the rate at which the crack tip is advancing (Andresen and Ford, 1994aa). DOE calculated the threshold stress intensity factor as a function of the Alloy 22 general corrosion rate and the repassivation slope, a parameter related to the rate at which Alloy 22 repassivates following a passive film rupture, as shown in SNL Eq. 19 (2007bb). DOE used a point value of 7.23 nm/yr [2.85×10^{-7} in] for the general corrosion rate of Alloy 22, on the basis of the 5-year corrosion data described in SAR Section 2.3.6.3. DOE determined the value of the repassivation slope by measuring the crack growth rate for fatigue precracked Alloy 22 compact tension specimens at 110 °C [230 °F] in basic simulated water and 150 °C [302 °F] in simulated concentrated water, as outlined in SAR Section 2.3.6.5.2.4 and SNL Section 6.4.4.2 (2007bb). For these conditions, the measured values of the repassivation slope are shown in SAR Table 2.3.6-17. DOE considered epistemic uncertainty in the repassivation slope and, in turn, the threshold stress intensity factor. In the TSPA code, DOE sampled the repassivation slope from a normal distribution. The mean threshold stress intensity factor was calculated as 6.62 $MPa\text{-}m^{0.5}$ [6.02 $ksi\text{-}in^{0.5}$], with lower and upper bounds of 1.96 $MPa\text{-}m^{0.5}$ [1.78 $ksi\text{-}in^{0.5}$] and 15.38 $MPa\text{-}m^{0.5}$ [14.0 $ksi\text{-}in^{0.5}$], respectively. DOE stated that this range corresponds to values reported in the technical literature (e.g., Jones, 1992aa) for other corrosion-resistant chromium-nickel-iron alloys.

DOE calculated the stress intensity factor profile for the waste package closure weld to determine whether the stress intensity factor could exceed the threshold value to cause crack propagation during the postclosure period, as described in SAR Section 2.3.6.5.2.3 and SNL Section 6.5 (2007bb). DOE used an approach in which it calculated the stress intensity factor for a relatively simple crack geometry and given stress distribution, then modified the solution for the waste package closure weld with a geometry correction factor, as detailed in SNL Section 6.5.3.3.3 (2007bb). DOE represented a radially oriented crack in the closure weld with the idealized case of a semicircular crack in an infinite plate, as outlined in SNL Section 6-17 (2007bb). The geometry correction factor was obtained by comparing the simplified solution to finite element analysis solutions for a number of crack sizes. Using this approach, DOE calculated the stress intensity factor profile for the plasticity-burnished waste package closure weld, as shown in SAR Table 2.3.6-16. Because DOE's calculated stress intensity factor was a linear function of the residual stress, uncertainty and variability in the residual stress profile were also represented in the stress intensity factor profile used in the TSPA code. SAR Figure 2.3.6-30 showed the angular variability in the stress intensity factor profile for the closure weld, and SAR Figure 2.3.6-32 showed the uncertainty in the stress intensity factor profile in the waste package closure weld.

Summary of NRC Staff's Review for Threshold Stress Intensive Fracture

The NRC staff reviewed the approach to calculate the threshold stress intensity factor for the waste package closure weld. The stress concentration at the tip of a crack generally decreases with increasing crack tip radius. In addition, the crack tip radius will increase if the crack sidewall general corrosion rate is higher than the crack tip advance rate. Therefore, the crack blunting criterion that DOE used is reasonable because it is consistent with the linear elastic fracture mechanics theory reported in the technical literature (e.g., Andresen and Ford, 1994aa). Further, the NRC staff notes DOE reasonably calculated the repassivation slope by measuring the crack growth rate for fatigue precracked Alloy 22 compact tension specimens because the crack growth rates were extremely low {less than 10^{-8} mm/s [3.94×10^{-10} in/s]} even though the stress intensity factor for the cracks was significantly higher than the sampled range for the threshold stress intensity factor in the TSPA code.

The NRC staff also verified the DOE calculation of the stress intensity factor profile for the waste package closure weld. DOE's modeling the radial crack in the waste package closure weld as a semicircular crack in an infinite plate is reasonable because the hoop stress in the weld region decreases rapidly with increasing distance from the weld line. Additionally, the variability and uncertainty in the stress intensity factor profile modeled in the TSPA code adequately reflects associated variability and uncertainty in DOE's calculated residual stress profile because the stress intensity factor profile in the TSPA code parameters were determined from mean, lower, and upper bounds of calculated residual stress profiles. Therefore, the stress intensity factor used in the TSPA code for the waste package outer barrier closure weld is reasonable, because it will not underestimate the stress intensity factor.

Crack Size and Density

The staff reviewed DOE's approaches to calculate the size and density of cracks initiated by fabrication-induced stresses and weld flaws, and those initiated by seismically induced stresses, respectively.

Cracks Initiated by Fabrication-Induced Stresses and Weld Flaws

DOE's analytic models assumed that the cracks initiated by fabrication-induced stresses exceeded the residual stress threshold and have a uniform size density. DOE assumed that it is energetically favorable for the cracks to have elliptical shape, as shown in SNL Figure 6.3-3 (2007aj), with crack length (i.e., major axis of the ellipse) of 50 mm [1.97 in]. DOE selected this length because it calculated that weld-induced stresses can persist on both sides of the weld centerline up to a distance approximately equal to the nominal thickness of the waste package outer barrier {i.e., 25 mm [0.98 in]}. The crack opening width was calculated using a fracture mechanics equation derived from an analysis of the energy associated with crack free surfaces. In this manner, DOE calculated a crack opening width of 0.1956 mm [7.70×10^{-3} in]. Given the crack length of 50 mm [1.97 in] and width of 0.1956 mm [7.70×10^{-3} in], DOE calculated that the opening area of an individual incipient crack was 7.682 mm^2 [1.19×10^{-2} in^2], which is assumed to be constant through the waste package wall (SNL, 2007aj). DOE stated that a crack of these dimensions would permit diffusive transport, but preclude advective transport by water [features, events, and processes (FEP) 2.1.03.10.0A; SNL (2008ac)]. Moreover, DOE assumed that the density of through-wall cracks is constrained by stress-field interactions in the area around the crack, which limit the ability of cracks in relative proximity to propagate through the waste package wall, as described in SNL Section 6.6.1 (2007bb). DOE's analysis indicated that the minimum spacing between through-wall cracks is equal to the thickness of the waste package outer barrier, which is 25 mm [0.98 in] (Structural Integrity Associates, 2002aa).

DOE used a different approach to model the size and density of weld flaw cracks in the waste package outer barrier closure weld. For the outer closure weld, DOE determined that the size and density of flaws would be small because of (i) highly controlled welding procedures that would limit flaw generation and (ii) extensive postweld nondestructive examination used to identify weld flaws, as detailed in SNL Section 6.3.1 (2007aa). DOE stated in SAR Section 1.5.2.7 that weld fabrication and inspection will follow the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Division 1, Subsection NC (American Society of Mechanical Engineers, 2001aa). The American Society of Mechanical Engineers (ASME) Code specifies that flaws larger than 1.6 mm [6.30×10^{-2} in] be detected and repaired, a requirement that is incorporated as a waste package Design Control Parameter [SAR Table 1.9-9, Design Control Parameter 03-17(b)]. To determine the size and density of flaws that may be expected in the waste package closure welds, DOE fabricated simulated welds (Smith, 2003aa; SAR Section 2.3.6.5.2.2). DOE stated that postweld nondestructive examination detected all flaws in the simulated welds than were larger than ASME Code allowed.

DOE performed a statistical analysis using data from the simulated welds to derive probability distributions for the size and density of flaws to be sampled in the TSPA analysis, as outlined in SNL Section 6.3.1 (2007aa). DOE used a Bayesian approach, consistent with weld flaw analyses in the technical literature (e.g., American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa). Following this approach, DOE determined that a Poisson distribution best represents the undetected weld flaw density. DOE calculated that after performing postweld nondestructive examination, the mean size of flaws that would go undetected is 1.0 mm [3.94×10^{-2} in], with 5th and 95th percentile values of 0.07 and 2.6 mm [2.76×10^{-3} and 1.02×10^{-1} in], respectively, as described in SNL Appendix A (2007aa). DOE calculated that after the nondestructive examination, there would be a mean of approximately one weld flaw per 140 m^3 [4.94×10^3 ft^3] of weld volume, with 5th and 95th percentile values of approximately one weld flaw per 56 m^3 [1.98×10^3 ft^3] and one weld flaw per 264 m^3 [9.32×10^3 ft^3], respectively, as detailed in SNL Appendix A (2007aa). Given the expected closure weld volume, DOE calculated that there is about 84 percent probability that a waste package will have no weld flaws, 14 percent probability that a waste package has one flaw, and 2 percent probability that a waste package has two or more flaws (SAR Table 2.3.6-18).

In the TSPA code, DOE calculated that only radially oriented flaws (i.e., those that make an angle of greater than 45° with respect to the weld line) are able to propagate because the primary stress component in the closure weld is the hoop stress (SAR Section 2.3.6.5.3.1). DOE determined that there is little driving force for the propagation of cracks that make an angle of less than 45° with respect to the primary stress direction, as outlined in SNL Section 6.3.4.3 (2007bb). DOE analyzed the flaws in the simulated welds to calculate a probability distribution for the orientation of flaws in the closure weld, as described in SNL Section 6.3.1.5 (2007aa). Using the Bayesian approach, DOE calculated that 0.8 percent of weld flaws will be radially oriented such that they can propagate under a hoop stress. DOE concluded that this calculation was supported by the independent analyses of Shcherbinskii and Myakishev (1970aa), who reported that most (~99 percent) weld flaws are oriented within about ±13° from the weld line.

In the TSPA code, DOE also assumed that only those weld flaws exposed to the environment by general corrosion during the postclosure period would be susceptible to propagation, as outlined in SNL Section 6.3.4.2 (2007bb). In the TSPA code, DOE calculated that 25 percent of weld flaws will be exposed and able to propagate based upon the approximate percentage of the waste package weld that would be removed by general corrosion during the postclosure period (SAR Section 2.3.6.3). On the basis of the small number of embedded weld flaws capable of propagation, DOE concluded that breach of the waste package outer barrier by weld flaw cracks is far less likely than breach by incipient cracks initiated where the residual stress is greater than the residual stress threshold, as outlined in SNL Section 6.3.5.1.3[a] (2008ag).

Summary of NRC Staff's Review for Cracks Initiated by Fabrication-Induced Stresses and Weld Flaws

The NRC staff reviewed the DOE approaches to calculate the size and density of cracks initiated by fabrication-induced stresses and weld flaws in the waste package outer barrier. Regarding cracks initiated by fabrication-induced stresses, the NRC staff notes that reports in the technical literature indicate that crack propagation requires energy to create the new crack surfaces (Anderson, 2005aa). Absent external stresses on the waste package, the NRC staff notes that the energy for crack propagation would come from the internal stresses in the weld. Thus, crack propagation will mitigate the residual stress in the weld by creating new free surfaces, thereby causing cracks to narrow as they propagate through the waste package outer barrier. DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model. Therefore, the opening area DOE calculated is reasonable because it does not underestimate the actual crack opening area. The NRC staff also notes that stress mitigation by crack propagation will constrain the density of through-wall cracks. As such, DOE's calculated crack density is reasonable because it does not underestimate the actual crack density.

The NRC staff also reviewed DOE's analysis of the size and density of flaws in the waste package closure weld. DOE's use of data from the simulated welds to evaluate the flaws in the waste package closure weld is reasonable because the simulated welds were fabricated with similar materials, procedures, equipment, and postwelding nondestructive evaluation methods as will be applied for the actual waste package welds. Further, DOE's use of a Bayesian approach to develop probability distributions for the size and density of undetected weld flaws is reasonable and consistent with NRC guidance (American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa) because this approach is appropriate when there is little information available regarding the characteristics of flaws in Alloy 22 welds. The NRC staff also notes that the DOE analysis was consistent with NRC analyses of flaws in dry storage cask welds, as described in NRC Appendix B (2006ab). Finally, DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model for weld flaws. On the basis of this information, the DOE distributions for weld flaw size and density used in the TSPA analyses are reasonable because these distributions will not underestimate these parameters.

Cracks Initiated by Seismically Induced Stresses

For stress corrosion cracks caused by impacts to the waste package during seismic ground motion, DOE sampled a parameter in the TSPA code called the crack area density, which is the product of the crack size and the crack density, as identified in SAR Section 2.3.4.5.1.4.1 and SNL Section 6.7.2 (2007bb). The crack area density is a unitless scalar measure that, when multiplied by the size of the damaged area on the waste package surface, gives the total open area of the through-wall cracks. For seismically induced stress corrosion cracking, DOE assumed that through-wall cracks have the same shape characteristics as cracks in the closure welds, as in SNL Figure 6.3-3 (2007aj). In contrast to the weld cracks, however, DOE considered uncertainty in the size and density for the cracks induced by seismic ground motion. DOE evaluated the uncertainty by calculating the crack area density using two conceptual models in which the crack size and crack density values were varied, as described in SNL Section 6.7.3 (2007bb). Both conceptual models use a regular hexagonal array of cracks on the waste package surface because this gives high effective crack density, as described in SNL Section 6.7.2 (2007bb). In the first conceptual model in SNL Section 6.7.3.1 (2007bb), cracks abutted tip-to-tip and the distance between parallel rows of cracks was the waste package wall thickness. In the second conceptual model in SNL Section 6.7.3.2 (2007bb), cracks could overlap (crack length was two times that in the first conceptual model) and the distance between crack centers was the wall thickness (crack number density was lower than that assumed in the first conceptual model). These resulted in SNL Eq.s 37 and 40 (2007bb) that DOE used to calculate the lower and upper bounds, respectively, for the crack area density sampled in the TSPA code.

Using this approach, DOE calculated a lower bound for crack area density of approximately 3.27×10^{-3} (i.e., stress corrosion cracking breached area is 0.327 percent of the waste package damaged area) and an upper bound of approximately 1.31×10^{-2} (i.e., stress corrosion cracking breached area is 1.31 percent of the waste package damaged area) (SAR Section 2.3.4.5.1.4.1). DOE considered an alternative conceptual model in which a single crack circumscribed the damaged area, as described in SNL Section 6.7.4 (2007bb). For this conceptual model, DOE calculated a crack area density of 7.22×10^{-3}, which is within the bounds given by the hexagonal crack network models. DOE determined that the alternative conceptual model provided support for the crack area density range calculated by the primary conceptual models. Therefore, in the TSPA code, DOE samples the crack area density from a uniform distribution between the bounding values given by the hexagonal crack network models.

Summary of NRC Staff's Review for Cracks Initiated by Seismically Induced Stresses

The NRC staff reviewed the DOE approach to establish the value of the crack area density for seismically induced stress corrosion cracks. The NRC staff notes that reports in the technical literature indicate that crack propagation requires energy to create new crack surfaces (Anderson, 2005aa). Absent external stresses on the waste package, the energy for crack propagation would necessarily come from the residual stresses generated from impacts to the waste package. Thus, crack propagation will mitigate the seismically induced stresses by creating new free surfaces, thereby causing cracks to narrow as they propagate through the waste package outer barrier. The NRC staff notes that DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model. Moreover, stress mitigation by crack propagation will constrain the density of through-wall cracks. As such, DOE's calculated range for the crack area density is reasonable because it does not underestimate the value of this parameter.

Abstraction and Integration

For the Nominal and Seismic Ground Motion Modeling Cases in the Total System Performance Assessment (TSPA) code, the model abstraction for stress corrosion cracking of the waste package closure weld was implemented in the Waste Package and Drip Shield Degradation Submodel, as detailed in SNL Section 6.3.5.1 (2008ag). In the submodel, the waste package closure weld area is represented by an annulus that is 1 patch wide and has the same radius as the waste package, as shown in SNL Figure 6.3.5-4 (2008ag). This results in about 40 patches to model the waste package closure weld. The waste package general corrosion abstraction and stress corrosion cracking abstraction, respectively, are implemented independently on each of the patches. As each patch thinned by general corrosion, the submodel calculated the residual stress on the patch on the basis of the through-wall residual stress profile. At each realization time step, the submodel compared the residual stress on the patch to the sampled residual stress threshold. If the residual stress on the patch was greater than the residual stress threshold, the submodel assumed that stress corrosion cracking initiated. The submodel also distributed weld flaws among the patches on the basis of the probability distributions for the weld flaw size and density. To determine whether initiated cracks in the waste package closure weld could propagate, the submodel calculated the stress intensity factor at the crack tip on the basis of the through-wall stress intensity factor profile, and compared it to the sampled threshold stress intensity factor. If the stress intensity factor was greater than the threshold stress intensity factor, the submodel assumed that the crack would propagate.
The crack growth rate was calculated using the slip-dissolution film-rupture model. A breached patch was assumed to have cracks with a size of 7.682 mm^2 [1.19×10^{-2} in^2] and spacing of 25 mm [0.98 in] (i.e., 6 cracks per patch). The output of the model was the time of waste package breach and the breach area. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

For the Nominal Modeling Case, DOE calculated that waste packages were not breached by stress corrosion cracking in the closure weld (i.e., less than 1 in 10^{-4} probability) for approximately 150,000 years after repository closure, and within 1 million years, a mean of approximately 50 percent of waste packages were breached [SAR Figure 2.1-10(a)]. Of those breached waste packages, DOE calculated that the mean fraction of breached area to total waste package surface area was less than 10^{-5} over 1 million years [SAR Figures 2.1-13(b) and 2.1-15(b)]. DOE calculated similar results for stress corrosion cracking of the closure weld in the Seismic Ground Motion Modeling Case, as shown in DOE Figures 1–4 (2009bj).

The model abstraction for stress corrosion cracking caused by impacts during seismic ground motion was implemented in the TSPA code in the Seismic Ground Motion Modeling Case, as outlined in SNL Section 6.6 (2008ag). In the Seismic Ground Motion Modeling Case, the residual stress threshold was sampled from a uniform distribution between 90 and 105 percent of the Alloy 22 at-temperature yield stress. Using the sampled residual stress threshold, DOE calculated the size of the waste package damaged area. The crack area density for the given damaged area was sampled from a uniform distribution, bounded by 3.27×10^{-3} and 1.31×10^{-2}. The product of the size of the damaged area and the crack area density gave the total open area of the stress corrosion cracking network. The output of the model was the time of waste package breach and the breach area. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

For stress corrosion cracking induced by impacts during seismic ground motion, DOE calculated that a mean of approximately 10 percent of commercial spent nuclear fuel waste packages were breached within about 250,000 years of repository closure and for the codisposal waste packages, a mean of approximately 40 percent were breached within about 150,000 years of repository closure (DOE, 2009cj). For both waste package types, the fraction of the waste package surface consisting of open cracks was less than 10^{-3}, as shown in DOE Figures 7 and 8 (2009cj). DOE stated that the response of the respective waste package types is different because commercial spent nuclear fuel waste packages are generally not damaged by seismic ground motion until breached by another mechanism (e.g., stress corrosion cracking of the closure weld) that leads to degradation of the waste package internals. Conversely, codisposal waste packages are structurally weaker and can be damaged by seismic ground motion regardless of previous damage. For both waste package types, the probability of seismically induced stress corrosion cracking plateaus within 250,000 years after repository closure because drip shields collapse and impinge the waste packages.

Summary of NRC Staff's Review for Abstraction and Integration for Stress Corrosion Cracking

The NRC staff reviewed the implementation and integration of the model abstraction for stress corrosion cracking of the waste package outer barrier in the postclosure performance assessment. DOE has provided information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the NRC staff notes that, with respect to the parameters used in the model abstraction, DOE adequately justified the ranges of these parameters and accounted for uncertainty in the model abstraction. Therefore, DOE's implementation of the model abstraction for stress corrosion cracking of the waste package outer barrier in the TSPA code is reasonable because it would not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Summary of Conclusions for Stress Corrosion Cracking of the Waste Package Outer Barrier

The NRC staff reviewed the DOE model abstraction for stress corrosion cracking of the waste package outer barrier that was implemented in the TSPA code. DOE used appropriate experimental tests and other independent technical literature to provide support for its model abstraction. Therefore, DOE reasonably accounted for stress corrosion cracking of the waste package outer barrier in the TSPA code.

2.2.1.3.1.3.2.4 Waste Package Early Failure

In SAR Section 2.2.2.3, DOE defined early failure to be a through-wall penetration of the waste package caused by manufacturing- and handling-induced defects, at a time earlier than would be expected for a nondefective waste package. DOE assumed that a waste package undergoes early failure if it is emplaced in the repository with an undetected manufacturing- or handling-induced defect. On the basis of the processes associated with waste package manufacturing and handling, DOE calculated that the probability of waste package early failure is best represented in the TSPA code by a lognormal distribution with a mean of 1.13×10^{-4} per waste package and an error factor of 8.17, as shown in SNL Table 6-7 (2007aa). The NRC staff reviewed the adequacy of this probability distribution in TER Section 2.2.1.2.2.4. This section addresses the NRC staff's review of the implementation of this probability distribution in the TSPA code.

Waste Package Early Failure Conceptual Model

DOE's conceptual model for early waste package failure is that the waste package with an undetected manufacturing- or handling-induced defect completely fails (i.e., is removed as a barrier to the flow of water) at the time of repository closure (SAR Section 2.3.6.6.1). DOE selected this representation because there are uncertainties associated with the timing and extent of breach for defective waste packages and a completely degraded waste package at the time of repository closure will not underestimate the timing and magnitude of radionuclide releases, as described in SNL Section 6.5.2 (2007aa). DOE concluded that this is a conservative representation of the early failure because the most likely consequence of improper waste package manufacturing or handling would be introduction of stress corrosion cracking, which tends to cause tight cracks that would limit the extent of radionuclide transport (SAR Section 2.3.6.6.4.1).

Summary of NRC Staff's Review for Waste Package Early Failure Conceptual Model

The NRC staff reviewed DOE's conceptual model for waste package early failure. The NRC staff notes that DOE attributed no barrier capability to the early failed waste package. The NRC staff noted, however, that consequences of manufacturing- or handling-induced defects (e.g., increased probability of stress corrosion cracking) would likely allow the waste package to maintain some barrier capability, which limits radionuclide releases. Because early failed waste packages in the DOE model have no barrier capability, the model will not cause DOE to underestimate the timing or magnitude of radionuclide releases. Therefore, the DOE conceptual model for waste package early failure analysis is reasonable.

Abstraction and Integration

The DOE model abstraction for early failure of the waste package was implemented in TSPA code in the Waste Package Early Failure Modeling Case, as detailed in SAR Section 2.4.2.1.5.2 and SNL Section 6.4.2 (2008ag). This modeling case uses most of same modeling components and submodels as were implemented in the Nominal Modeling Case. In the Nominal Modeling Case, however, the Waste Package and Drip Shield Degradation Submodel provides the waste package and drip shield breached areas as a function of time to the Engineered Barrier System Flow and Transport Submodels and the Waste Form Degradation and Mobilization Model

Components. In the Waste Package Early Failure Modeling Case, the Waste Package and Drip Shield Degradation Submodel was replaced with the Waste Package Early Failure model, which simulated early failure by treating all patches on the failed waste package as breached at the time of repository closure.

DOE projected that the dose consequence of a waste package early failure would depend primarily upon the type of waste package, the environmental conditions at the waste package (e.g., temperature and relative humidity), and whether the waste package was in seepage conditions. Therefore, the Waste Package Early Failure Modeling Case calculated the dose consequence of a single early failure of a commercial spent nuclear fuel and codisposal waste package in each of the five percolation subregions with and without seepage conditions. The TSPA code then calculated the expected dose using the early failure probability [sampled from the distribution given in SNL Table 7-1 (2007aa)], the distribution for the waste package type, and the seepage fraction for each percolation bin.

DOE calculated that there is approximately 55.8 percent probability of no waste package early failures, approximately 22.4 percent probability of one early failure, approximately 9.6 percent probability of two early failures, and approximately 12.3 percent probability of three or more early failures in the repository, as outlined in SNL Table 6.4-1 (2008ag). Waste package early failure makes a small contribution to DOE’s calculated mean annual dose within approximately 20,000 years following closure {less than 0.001 mSv [0.1 mrem]}, with a declining contribution thereafter (SAR Figure 2.4-18).

Summary of NRC Staff’s Review for Abstraction and Integration of the Waste Package Early Failure Model

The NRC staff reviewed the implementation of the waste package early failure model in the TSPA code. DOE has provided information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that the model abstraction was consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, with respect to the parameters used in the model abstraction, DOE’s parameter ranges and accounting of uncertainty in the model abstraction are reasonable. Therefore, DOE’s implementation of the Waste Package Early Failure model abstraction in the TSPA code is reasonable because it would not underestimate the timing or magnitude of radionuclide release to the accessible environment.

NRC Summary of Conclusions for Waste Package Early Failure

The NRC staff reviewed the DOE model abstraction for early failure of the waste package outer barrier that was implemented in the TSPA code. DOE provided appropriate support for the model abstraction. Therefore, DOE reasonably accounted for waste package early failure in the TSPA analysis.

2.2.1.3.1.4 NRC Staff Conclusions

The NRC staff notes that the DOE description of this model abstraction for the degradation of engineered barriers is consistent with the guidance in the YMRP. The NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.1.5 References

Ahn, T., H. Jung, X. He, and O. Pensado. 2008aa. "Understanding Long-Term Corrosion of Alloy 22 Container in the Potential Yucca Mountain Repository for High-Level Nuclear Waste Disposal." *Journal of Nuclear Materials.* Vol. 379. pp. 33–41.

American Nuclear Society/Institute of Electrical and Electronics Engineers. 1983aa. NUREG/CR–2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants." Vols. 1 and 2. Washington, DC: NRC.

American Society of Mechanical Engineers. 2001aa. *ASME Boiler and Pressure Vessel Code.* New York City, New York: American Society of Mechanical Engineers.

American Society of Mechanical Engineers. 1969aa. *Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2.* New York City, New York: American Society of Mechanical Engineers.

Amy, P.S., C. Pantle, T.A. Else, and C. Neuwohner. 2002aa. "Humidity and Temperature Boundaries for Biofilm Formation in Yucca Mountain." TR–02–002. Rev. 0. LSN DN200000748. Las Vegas, Nevada: University of Nevada.

Anderson, T.L. 2005aa. *Fracture Mechanics*. 3rd Edition. Boca Raton, Florida: Taylor and Francis.

Andresen, P.L. 1991aa. "Fracture Mechanics Data and Modeling of Environmental Cracking of Nickel-Base Alloys in High Temperature Water." Proceedings of the CORROSION 91 Conference. Paper No. 44. Houston, Texas: NACE International.

Andresen, P.L. and F.L. Ford. 1994aa. "Fundamental Modeling of Environment Cracking for Improved Design and Lifetime Evaluation in BWRs." *International Journal of Pressure Vessels and Piping.* Vol. 59, Nos. 1–3. pp. 61–70.

Andresen, P.L. and Y.J. Kim. 2009aa. "Final Report for October 1, 2007–September 30, 2008: Stress Corrosion Crack Initiation and Growth Measurements in Environments Relevant to High-Level Nuclear Waste Packages." GE–GRC–Sandia–2008–1. DTN: MO0902SCCIGMFR.000. Niskayuna, New York: General Electric Global Research Center.

Andresen, P.L. and Y.J. Kim. 2007aa. "Stress Corrosion Crack Initiation and Growth Measurements in Environments Relevant to High-Level Nuclear Waste packages: Final Report for October 1, 2006–September 30, 2007." GE–GRC–Sandia–2007–3. Niskayuna, New York: General Electric Global Research Center.

Andresen, P.L. and Y.J. Kim. 2006aa. "Stress Corrosion Crack Initiation and Growth Measurements in Environments Relevant to High-Level Nuclear Waste packages." GE–GRC–Bechtel–2006–2. MOL.20061109.0070/MOL.20061108.0004. Niskayuna, New York: General Electric Global Research Center.

Ashida, Y., L. G. McMillion, and M. Misra. 2008aa. "Communication C: A Heated Electrode Test System for Studying Corrosion Behavior of Alloy 22." *Nuclear Waste Research: Siting, Technology and Treatment*. A.P. Lattefer, ed. Hauppauge, New York: Nova Science Publishers, Inc. pp. 25–36.

Ashida. Y., L.G. McMillion, and M.L. Taylor. 2007aa. "The Effect of Temperature Oscillation on the Passive Corrosion Properties of Alloy 22." *Electrochemistry Communications*. Vol. 9, Issue 5. pp. 1,102–1,106.

ASM International. 1993aa. "Welding, Brazing, and Soldering." Vol. 6. *Metals Handbook*. Metals Park, Ohio: ASM International.

ASM International. 1987aa. "Corrosion". Vol. 13. *Metals Handbook.* 9th Edition. Metals Park, Ohio: ASM International.

ASTM International. 2007aa. "Standard Practice for Maintaining Constant Relative Humidity by Means of Aqueous Solutions." ASTM E104–02. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 2003aa. "Standard Test Method for Conducting Cyclic Potentiodynamic Polarization Measurements for Localized Corrosion Susceptibility of Iron-, Nickel-, or Cobalt-Based Alloys." ASTM G 61-86. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 2003ab. "Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens." ASTM G 1-03. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 2000aa. "Standard Practice for Preparation and Use of Direct Tension Stress-Corrosion Test Specimens." ASTM G 49-85. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 2000ab. "Standard Practice for Slow Strain Rate Testing to Evaluate the Susceptibility of Metallic Materials to Environmentally Assisted Cracking." ASTM G 129-00. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 1999aa. "Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens." ASTM G 1-90. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 1998aa. "Standard Test Method for Conducting Potentiodynamic Polarization Resistance Measurements." ASTM G 59-97. West Conshohocken, Pennsylvania: ASTM International.

ASTM International. 1994aa. "Standard Practice for Making and Using U-Bend Stress-Corrosion Test Specimens." ASTM G 30-94. Philadelphia, Pennsylvania: ASTM International.

ASTM International. 1991aa. "Standard Test Method for Plane-Strain Fracture Toughness of Metallic Materials." ASTM E 399-90. West Conshohocken, Pennsylvania: ASTM International.

Aylor, D.M., R.A. Haysz, R.M. Kain, and R.J. Ferrara. 1999aa. “Crevice Corrosion Performance of Candidate Naval Ship Seawater Valve Materials in Quiescent and Flowing Natural Seawater.” Proceedings of the CORROSION/99 Conference. Paper No. 329. Houston, Texas: NACE International.

Brossia, C.S. and G.A. Cragnolino. 2004aa. “Effect of Palladium on the Corrosion Behavior of Titanium.” *Corrosion Science*. Vol. 46. pp. 1,693–1,711.

Brossia, C.S. and G.A. Cragnolino. 2001ab. “Effect of Palladium on the Localized and Passive Dissolution of Titanium.” Proceedings of the Corrosion/2001 Conference. Paper No. 01127. Houston, Texas: NACE International.

Brossia, C.S. and G.A. Cragnolino. 2000aa. “Effects of Environmental, Electrochemical, and Metallurgical Variables on the Passive and Localized Dissolution of Ti Grate 7.” CORROSION/2000. Paper No. 00211. Houston, Texas: NACE International.

Brossia, C.S., L. Browning, D.S. Dunn, O.C. Moghissi, O. Pensado, and L. Yang. 2001aa. “Effect of Environment on the Corrosion of Waste package and Drip Shield Materials.” CNWRA 2001-003. San Antonio, Texas: CNWRA.

Brown, A.D. 1976aa. “Microbial Water Stress.” *Bacteriological Reviews.* Vol. 40. pp. 803–846.

BSC. 2007bu. “Yucca Mountain Project Engineering Specification Prototype Drip Shield.” 000–3SS–SSE0–00100–000. Rev. 000. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004as. “General Corrosion and Localized Corrosion of the Drip Shield.” ANL–EBS–MD–000004. Rev. 02. AD 01, ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Chiang, K.-T., D.S. Dunn, and G.A. Cragnolino. 2006aa. “Combined Effect of Bicarbonate and Chloride Ions on Stress Corrosion Cracking Susceptibility of Alloy 22.” Proceedings of the CORROSION/2006 Conference. Paper No. 06506. Houston, Texas: NACE International.

Chiang, K.-T., D.S. Dunn, and G.A. Cragnolino. 2005aa. “Effect of Groundwater Chemistry on Stress Corrosion Cracking.” Proceedings of the CORROSION/2005 Conference. Paper No. 05463. Houston, Texas: NACE International.

Codell, R.B. and B.W. Leslie. 2006aa “Drip Shield Corrosion from Fluoride in Dents.” Proceedings of 11th International High-Level Radioactive Waste Management Conference, Las Vegas, American Nuclear Society.

Costa, D. and P. Marcus. 1993aa. “Modification of Passive Films Formed on Ni-Cr-Fe Alloys With Chromium Content in the Alloy and Effects of Adsorbed or Segregated Sulphur.” Proceedings of the European Symposium on Modifications of Passive Film, Paris, France, February 15–17, 1993. P. Marcus, B. Barous, and M. Keddam, eds. London, England: Institute of Materials, Minerals, and Mining. pp. 17–25.

Cragnolino, G.A., D.S. Dunn, C.S. Brossia, V. Jain, and K.S. Chan. 1999aa. "Assessment of Performance Issues Related to Alternate Engineering Barrier System Materials and Design Options." CNWRA 99-003. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

Dixit, S., S. Roberts, K. Evans, T. Wolery, and S. Carroll. 2006aa. "General Corrosion and Passive Film Stability–FY05 Summary Report." UCRL–TR–217393. Livermore, California: Lawrence Livermore National Laboratory.

DOE. 2010ae. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 2." Letter (February 22) J.R. Williams to J.H. Sulima (NRC). ML100540266. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.4) Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML092110472. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 3." Letter (May 7) J.R. Williams to J.H. Sulima (NRC). ML091280184. ML091280185. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cl. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 2." Letter (April 13) J.R. Williams to J.H. Sulima (NRC). ML091100634. Washington, DC: DOE, Office of Technical Management.

DOE. 2009dg. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 2." Letter (April 8) J.R. Williams to J.H. Sulima (NRC). ML090980537. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cm. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 4." Letter (September 10) J.R. Williams to J.H. Sulima (NRC). ML092540339. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cn. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 1." Letter (March 25) J.R. Williams to J.H. Sulima (NRC). ML090840553. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Dunn, D.S., Y.-M. Pan, X. He, L.T. Yang, and R.T. Pabalan. 2006ab. “Evolution of Chemistry and Its Effects on the Corrosion of Engineered Barrier Materials.” The 30^{th} Symposium on the Scientific Basis for Nuclear Waste Management Materials Research Society 2006 Fall Meeting, Boston, Massachusetts, November 27–December 1, 2006. Pittsburgh, Pennsylvania: Materials Research Society.

Dunn, D.S., O. Pensado, Y.-M. Pan, R.T. Pabalan, L. Yang, X. He, and K.T. Chiang. 2005aa. “Passive and Localized Corrosion of Alloy 22—Modeling and Experiments.” Rev. 01. CNWRA 2005-002. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

Dunn, D.S., G.A. Cragnolino, and N. Sridhar. 2000aa. “An Electrochemical Approach to Predicting Long-Term Localized Corrosion of Corrosion-Resistant High-Level Waste Container Materials.” *Corrosion*. Vol. 56. pp. 90–104.

Elibiache, A. and P. Marcus. 1992aa. “The Role of Molybdenum in the Dissolution and the Passivation of Stainless Steels With Adsorbed Sulphur.” *Corrosion Science*. Vol. 33, No. 2. pp. 261–269.

Else, T.A., C.R. Pantle, and P.S. Amy. 2003aa. “Boundaries for Biofilm Formation: Humidity and Temperature.” *Applied and Environmental Microbiology*. Vol. 69, No. 8. pp. 5,006–5,010.

Evans, K.J., A. Yilmaz, S.D. Day, L.L. Wong, J.C. Estill, and R. Rebak. 2005aa. “Using Electrochemical Methods to Determine Alloy 22's Crevice Corrosion Repassivation Potential.” *Journal of Metals.* Vol. 57, No. 1. pp. 56–61.

Evans, K.J., M.L. Stuart, R.A. Etien, G.A. Hust, J.C. Estill, and R.B. Rebek. 2005ab. “Long-Term Corrosion Potential and Corrosion Rate of Creviced Alloy 22 in Chloride Plus Nitrate Brines.” UCRL–CONF–216910. Livermore, California: Lawrence Livermore National Laboratory.

Fix, D.V., J.C. Estill, G. A. Hust, K.J. King, S.D. Day, and R.B. Rebak. 2003aa. “Influence of Environmental Variables on the Susceptibility of Alloy 22 to Environmentally Assisted Cracking.” Proceedings of the CORROSION 2003/Conference. Paper No. 03542. Houston, Texas: NACE International.

Fontana, M.G. and N.D. Greene. 1978aa. *Corrosion Engineering*. 2nd Edition. New York City, New York: McGraw Hill.

Ford, F.P. and P.L. Andresen. 1988aa. “Development and Use of a Predictive Model of Crack Propagation in 304/316L, A533B/A508, and Inconel 600/182 Alloys in 288˚C Water.” Proceedings of the Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors, Traverse City, Michigan, August 30–September 3, 1988. G.J. Theus and J.R. Weeks, eds. Warrendale, Pennsylvania: The Metallurgical Society. pp. 789–800.

Gray, J.J., J.R. Hayes, G.E. Gdowski, B.E. Viani, and C.A. Orme. 2006aa. “Inhibiting Effects of Nitrates on the Passive Film Breakdown of Alloy 22 in Chloride Environments.” *Journal of the Electrochemical Society*. Vol. 153, No. 5. pp. B156–B161.

Greenspan, L. 1977aa. “Humidity Fixed Points of Binary Saturated Aqueous Solutions.” *Journal of Research of the National Bureau of Standards—A: Physics and Chemistry.* Vol. 81A. pp. 89–96.

Haynes International. 1997aa. “Hastelloy Alloy C-276.” Kokomo, Indiana: Haynes International.

Haynes International. 1997ab. “Hastelloy C-22 Alloy.” Kokomo, Indiana: Haynes International.

He, X. and D.S. Dunn. 2005aa. “Crevice Corrosion Penetration Rates of Alloy 22 in Chloride-Containing Waters–Progress Report.” CNWRA 2006-001. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

He, X., B. Brettmann, and H. Jung. 2009aa. “Effects of Test Methods on Crevice Corrosion Repassivation Potential Measurements of Alloy 22.” *Corrosion.* Vol. 65. pp. 449–460.

He, X., J.J. Noel, and D.W. Shoesmith. 2007aa. "Temperature Effects on Oxide Film Properties of Grade-7 Titanium." *Corrosion.* Vol. 63. pp. 781–792.

He, X, D.S. Dunn, and A.A. Csontos. 2007ab. “Corrosion of Similar and Dissimilar Metal Crevices in the Engineered Barrier System of a Potential Nuclear Waste Repository.” *Electrochimica Acta*. Vol. 52. pp. 7,556–7,569.

Horn, J., S. Martin, C. Carrillo, and T. Lian. 2005aa. “Microbial Effects on Nuclear Waste Packaging Materials.” UCRL–TR-213915. Rev 1. Livermore, California: Lawrence Livermore National Laboratory.

Hua, F. and G. Gordon. 2004aa. “Corrosion Behavior of Alloy 22 and Ti Grade 7 in a Nuclear Waste Repository Environment.” *Corrosion.* Vol. 60, No. 8. pp. 764–777.

Hunkeler, F. and H. Boehni. 1983aa. “Pit Growth Measurements on Stainless Steels.” Passivity of Metals and Semiconductors, Proceedings of the Fifth International Symposium on Passivity, Bombannes, France, May 30–June 3, 1983. M. Froment, ed. New York City, New York: Elsevier. pp. 655–660.

Hur, D.H. and Y.S. Park. 2006aa. “Effect of Temperature on the Pitting Behavior and Passive Film Characteristics of Alloy 600 in Chloride Solution.” *Corrosion*. Vol. 62, No. 9. pp. 745–750.

Ishikawa, H., A. Honda, and N. Sasaki. 1994aa. “Long Life Prediction of Carbon Steel Overpacks for Geological Isolation of High-Level Radioactive Waste.” Life Prediction of Corrodible Structures. R.N. Parkins, ed. Houston, Texas: NACE International. Vol. 1. pp. 454–483.

Jones, D.A. 1996aa. *Principles and Prevention of Corrosion*. 2nd Edition. Upper Saddle River, New Jersey: Prentice Hall.

Jones, R.H., ed. 1992aa. “Stress-Corrosion Cracking.” Materials Park, Ohio: ASM International.

Jones, R.H. and R.E. Ricker. 1987aa. “Stress-Corrosion Cracking.” *Metals Handbook, Vol. 13: Corrosion—9th Edition.* Metals Park, Ohio: ASM International. pp. 145–163.

Jung, H. 2010aa. “A Review for Long-Term Persistence of Passive Film of Alloy 22 Under Potential YM Repository Environments.” Scientific Notebook No. 835. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

Jung, H., T. Mintz, L. Yang and T. Ahn. 2008aa. “Long-Term Persistence of the Passive Film on Alloy 22 at Elevated Temperatures in the Potential Yucca Mountain Repository Environment.” ML090220175 and ML090220186. Washington, DC: NRC.

Jung, H., T. Mintz, D.S. Dunn, O. Pensado, and T. Ahn. 2007aa. “A Review of the Long-Term Persistence of the Passive Film on Alloy 22 in Potential Yucca Mountain Repository Environments.” ML072880595. Washington, DC: NRC.

Lee, S.G. and A.A. Solomon. 2006aa. “Localized Corrosion of Alloy C22 Nuclear Waste Canister Material Under Limiting Conditions.” *Materials Science and Engineering.* Vol. A 434. pp. 114–123.

Lian, T., M.T. Whalen, and L.L. Wong. 2005aa. “Effects of Oxide Film on the Corrosion Resistance of Titanium Grade 7 in Fluoride-Containing NaCl Brines.” Proceedings of the Corrosion/2005, 60th Annual Conference and Exposition, 1945–2005, April 3-7, 2005. Paper No. 05609. Houston, Texas: NACE International.

Lian, T., S. Martin, D. Jones, A. Rivera, and J. Horn. 1999aa. “Corrosion of Candidate Container Materials by Yucca Mountain Bacteria.” Proceedings of the CORROSION/1999 Conference. Paper No. 476. Houston, Texas: NACE International.

Lin C., B. Leslie, R. Codell, H. Arlt, and T. Ahn. 2003aa. “Potential Importance of Fluoride to Performance of the Drip Shield.” Proceedings of the American Nuclear Society 10th International High-Level Radioactive Waste Management Conference, March 30–April 2, 2003, Las Vegas, Nevada.

Lloyd, A.C., J.J. Noël, S. McIntyre, and S.W. Shoesmith. 2004aa. “Cr, Mo and W Alloying Additions In Ni and Their Effect on Passivity.” *Electrochimica Acta*. Vol. 49. pp. 3,015–3,027.

Lloyd, A.C., D.W. Shoesmith, N.S. McIntyre, and J.J. Noel. 2003aa. “Effects of Temperature and Pote+B25ntial on the Passive Corrosion Properties of Alloys C22 and C276.” *Journal of the Electrochemical Society.* Vol. 150. pp. B120–130.

Lorenzo de Mele, M.F. and M.C. Cortizo. 2000aa. “Electrochemical Behavior of Titanium in Fluoride-Containing Saliva.” *Journal of Applied Electrochemistry*. Vol. 30. pp. 95–100.

Macdonald, D.D. and M. Urquidi-Macdonald. 1991aa. “A Coupled Environment Model for Stress Corrosion Cracking in Sensitized Type 304 Stainless Steel in LWR Environments.” *Corrosion Science.* Vol. B33 32, No. 1. pp. 51–81.

Macdonald, D.D., M. Urquidi-Macdonald, and P.-C. Lu. 1994aa. “The Coupled Environmental Fractural Model—A Deterministic Method for Calculating Crack Growth Rates.” Proceedings of the CORROSION/1994 Conference. Paper No. 246. Houston, Texas: NACE International.

Marcus, P. 2001aa. "Long-Term Extrapolation of Passive Behavior: Proceedings From an International Workshop on Long-Term Extrapolation of Passive Behavior." Presented at the Nuclear Waste Technical Review Board Meeting, Arlington, Virginia, July 19–20, 2001. Paris, France: Universite Pierre et Marie Curie. pp. 55–60.

Marcus, P. and J.M. Grimal. 1990aa. "The Antagonistic Roles of Chromium and Sulfur in the Passivation of Ni-Cr-Fe Alloys Studies by XPS and Radiochemical Techniques." *Corrosion Science*. Vol. 31. pp. 377–382.

Marcus, P. and M. Moscatelli. 1989aa. "The Role of Alloyed Molybdenum in the Dissolution and the Passivation of Nickel-Molybdenum Alloys in the Presence of Adsorbed Sulfur." *Journal of the Electrochemical Society*. Vol. 136, No. 6. pp. 1,634–1,637.

Marcus, P. and H. Talah. 1989aa. "The Sulfur Induced Breakdown of the Passive Film and Pitting Studied on Nickel and Nickel Alloys." *Corrosion Science*. Vol. 29. pp. 455–463.

Marcus, P., H. Talah, and J. Oudar. 1988aa. "Breakdown of the Passive Film on Nickel and Nickel Alloys Induced by Sulfur." *Key Engineering Materials*. Vols. 20–28, Issue 4. pp. 3,947–3,952.

Marcus, P., A. Testier, and J. Oudar. 1984aa. "The Influence of Sulphur on the Dissolution and the Passivation of a Nickel-Iron Alloy—Electrochemical and Radiotracer Measurements." *Corrosion Science*. Vol. 24, No.4. pp. 259–268.

Marcus, P., I. Olefjord, and J. Oudar. 1984ab. "The Influence of Sulfur on the Dissolution and the Passivation of a Nickel-Iron Alloy—Sulfur Analysis by ESCA." *Corrosion Science*. Vol. 24, No. 4. pp. 269–278.

Marcus, P., J. Oudar, and I. Olefjord. 1980aa. "Studies of the Influence of Sulfur on the Passivation of Nickel by Auger Electron Spectroscopy and Electron Spectroscopy for Chemical Analysis." *Materials Science and Engineering*. Vol. 42. pp. 191–197.

Marsh, G.P., K.J. Taylor, and A.H. Harker. 1991aa. "The Kinetics of Pitting Corrosion of Carbon Steel Applied to Evaluating Containers for Nuclear Waste Disposal." SKB TR–91–62. Stockholm, Sweden: Svensk Kärnbränsleförsörjning AB.

Martin, S., J. Horn, and C. Carrillo. 2004aa. "Micron-Scale MIC of Alloy-22 After Long-Term Incubation in Saturated Nuclear Waste Repository Microcosms." Proceedings of the CORROSION/2004 Conference. Paper No. 04596. Houston, Texas: NACE International.

Mattsson, H. and I. Olefjord. 1990aa. "Analysis of Oxide Formed on Ti During Exposure in Bentonite Clay-I. The Oxide Growth." *Werkstoffe und Corrosion.* Vol. 41, No. 7. Weinheim, Germany: VCH Verlagsgesellschaft mbH. pp. 383–390.

McMillion, L.G., A. Sun, D.D. Macdonald, and D.A. Jones. 2005aa. "General Corrosion of Alloy 22: Experimental Determination of Model Parameters From Electrochemical Impedance Spectroscopy Data." *Metallurgical and Materials Transactions A.* Vol. 36A. pp. 1,129–1,141.

Mintz, T.M. and T.M. Devine. 2004aa. "Influence of Surface Films on the Susceptibility of Inconel 600 to Stress Corrosion Cracking." *Key Engineering Materials.* Vols. 261–263. pp. 875–884.

Mintz, T. and X. He. 2009aa. "Modeling of Hydrogen Uphill Diffusion in Dissimilar Titanium Welds." Proceedings of the CORROSION/2009 Conference. Paper No. 09430. Houston, Texas: NACE International.

Miserque, F., B. Huet, G. Azou, D. Bendjaballah, and V. L'Hostis. 2006aa. "X-Ray Photoelectron Spectroscopy and Electrochemical Studies of Mild Steel FeE500 Passivation in Concrete Simulated Water." *Journal de Physique IV Proceedings.* Vol. 136. pp. 89–97.

Mohr, W.C. 1996aa. "Internal Surface Residual Stresses in Girth Butt-Welded Steel Pipes." *Residual Stresses in Design, Fabrication, Assessment and Repair, PVP.* Vol. 321. New York City, New York: American Society of Mechanical Engineers. pp. 37–44.

Montemor, M.F., M.G.S. Ferreira, M. Walls, B. Rondot, and M. Cunha Belo. 2003aa. "Influence of pH on Properties of Oxide Films Formed on Type 316L Stainless Steel, Alloy 600, and Alloy 690 in High-Temperature Aqueous Environments." *Corrosion*. Vol. 59, No. 1. pp. 11–21.

Mughabghab, S.F. and T.M. Sullivan. 1989aa. "Evaluation of the Pitting Corrosion of Carbon Steels and Other Ferrous Metals in Soil Systems." *Waste Management.* Vol. 9, No. 4. Elmsford, New York: Pergamon Press. pp. 239–251.

Mulford, S.J. and D. Tromans. 1988aa. "Crevice Corrosion of Nickel-Based Alloys in Neutral Chloride and Thiosulfate Solutions." *Corrosion*. Vol. 44, No. 12. pp. 891–900.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2006ab. NUREG–1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant." Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC. 1977aa. NUREG–0376, "Residual Stresses at Girth-Butt Welds in Pipes and Pressure Vessels." Rev. 5. Washington, DC: NRC.

Oka, Y.I., S. Mihara, and H. Miyata. 2007aa. "Effective Parameters for Erosion Caused by Water Droplet Impingement and Applications to Surface Treatment Technology." *Wear.* Vol. 263. pp. 386–394.

Orme, C.A. 2005aa. "The Passive Film on Alloy 22." UCRL–TR–215277. Livermore, California: Lawrence Livermore National Laboratory.

Pabalan, R.T. 2010aa. "Quantity and Chemistry of Water Contacting Engineered Barriers Integrated Subissue." Electronic Scientific Notebook 930E. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

Pasupathi, V. 2000aa. "Documentation of Literature on Residual Stress Measurements." Interoffice Correspondence (May 19) to G.M. Gordon. LV.WASTE PACKAGE.VP.05/00-070, With Enclosures. LSN DEN001312636. Las Vegas, Nevada: CRWMS M&O.

Pedersen, K. and F. Karlsson. 1995aa. "Investigation of Subterranean Microorganisms: Their Importance for performance Assessment of Radioactive Waste Disposal." SKB 95-10. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Company.

Pensado, O., D.S. Dunn, G.A. Cragnolino, and V. Jain. 2002aa. "Passive Dissolution of Container Materials—Modeling and Experiments." CNWRA 2003-01. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses.

Pourbaix, M. 1974aa. *Atlas of Electrochemical Equilibria in Aqueous Solutions*. Houston, Texas: NACE International.

Prevey, P.S. and J. Cammett,. 2001aa. "Low Cost Corrosion Damage Mitigation and Improved Fatigue Performance of Low-Plasticity Burnished 7075-T6." *Journal of Materials Engineering and Performance.* Vol. 10. pp. 548–555.

Pulvirenti, A.L., K.M. Needham, D.S. Wong, M.A. Adel-Hadadi, A. Barkatt, C.R. Marks, and J.A. Gorman. 2003aa. "Fluoride Corrosion of Ti-Grade 7: Effects of Other Ions." CORROSION/2003, 58th Annual Conference and Exposition, San Diego, California, March 16–20, 2003. Paper No. 03686. Houston, Texas: NACE International.

Pulvirenti, A.L., K.M. Needham, M.A. Adel-Hadadi, A. Barkatt, C.R. Marks, and J.A. Gorman. 2002aa. "Corrosion of Titanium Grade 7 in Solutions Containing Fluoride and Chloride Salts." CORROSION/2002, 57th Annual Conference and Exposition, Denver, Colorado, April 7–11, 2002. Paper No. 02552. Houston, Texas: NACE International.

Rebak, R.B. 2005aa. "Factors Affecting the Crevice Corrosion Susceptibility of Alloy 22." Proceedings of the CORROSION/2005 Conference. Paper No. 05610. Houston, Texas: NACE International.

Rybicki, E.F. and R.B. Stonesifer. 1979aa. "Computation of Residual Stresses Due to Multipass Welds in Piping Systems." *Journal of Pressure Vessel Technology.* Vol. 101. pp. 149–154.

Sala, B., S. Chevalier, A. Gelpi, H. Takenouti, and M. Keddam. 1999aa. "Electrochemical Study of the Corrosion Processes of the Secondary Side of Steam Generators." Proceedings of the 9th International Conference on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors. S.M. Bruemmer, F.P. Ford, and G.S. Was, eds. Warrendale, Pennsylvania: Materials Research Society. pp. 587–598.

Sala, B., S. Chevalier, M. Organista, A. Gelpi, A. Stutzmann, and M. Dupin. 1998aa. "Study of the Corrosion Process of the Secondary Side of Steam Generators: An Electrochemical Study," JAIF International Conference on Water Chemistry in Nuclear Power Plants, Water Chemistry '98. Kashiwazaki City, Niigata: Japan Atomic Industrial Forum. p. 498–504.

Sala, B., P. Combrade, A. Gelpi, and M. Dupin. 1996aa. "The Use of Tube Examinations and Laboratory Simulations to Improve the Knowledge of Local Environments and Surface Reactions in TSPs." *Control of Corrosion on the Secondary Side of Steam Generators.*

R.W. Staehle, J.A. Gorman, and A.R. McIlree, eds. Houston, Texas: NACE International. pp. 483–497.

Sala, B., R. Erre, and A. Gelpi. 1993aa. “Local Chemistry and Formation of Deposits on the Secondary Side of Steam Generators.” Proceedings of the 6th International Conference on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors. R.E. Gold and E.P. Simonen, eds. Warrendale, Pennsylvania: Materials Research Society. pp. 215–226.

Schutz, R.W. 2005aa. “Corrosion of Titanium and Titanium Alloys.” *ASM Handbook—Vol. 13B: Corrosion*. Materials. Materials Park, Ohio: ASM International.

Schutz, R.W. and J.S. Grauman. 1986aa. “Corrosion Behavior of Titanium and Other Alloys in Laboratory FGD Scrubber Environments.” *Materials Performance.* Vol. *25*, No. 4. Houston, Texas: NACE International. pp. 35–42.

Schutz, R.W. and D.E. Thomas. 1987aa. “Corrosion of Titanium and Titanium Alloys.” *ASM Handbook—Corrosion: Vol. 13.* Materials Park, Ohio: ASM International. Pp. 669–706.

Scully, J.R., G. Ilevbare, and C. Marks. 2001aa. “Passivity and Passive Corrosion of Alloys 625 and 22.” SEAS Report No. UVA/527653/MSE01/103. Charlottesville, Virginia: University of Virginia, School of Engineering & Applied Science.

Shack, W.J., W.A. Ellington, and L.E. Pahis. 1980aa. “Measurement of Residual Stresses in Type 304 Stainless Steel Butt Weldments.” EPRI NP–1413. Palo Alto, California: Electric Power Research Institute.

Shan, X. and J. H. Payer. 2007aa. “Comparison of Ceramic and Polymer Crevice Formers on the Crevice Corrosion Behavior of Ni-Cr-Mo Alloy C-22.” Proceedings of the CORROSION/2007 Conference. Paper No. 07582. Houston, Texas: NACE International.

Sharland, S.M., C.C. Naish, K.J. Taylor, and G.P. Marsh. 1994aa. “An Experimental and Modelling Study of the Localized Corrosion of Carbon Steel Overpacks for the Geological Disposal of Radioactive Waste.” *Life Prediction of Corrodible Structures*. R.N. Parkins, ed. Houston, Texas: NACE International. pp. 402–418.

Shcherbinskii, V.G. and V.M. Myakishev. 1970aa. “Statistical Distribution of Welding Defects with Respect to Azimuth.” Translated from Defektoskopiya, No. 4. New York City, New York: Plenum Publishing. pp. 143–144.

Shukla, P.K., D.S. Dunn, K.-T. Chiang, and O. Pensado. 2006aa. “Stress Corrosion Cracking Model for Alloy 22 In the Potential Yucca Mountain Repository Environment.” Proceedings of the CORROSION/2006 Conference. Paper No. 06502. Houston, Texas: NACE International.

Smailos, E. 1993aa. “Corrosion of High-Level Waste Packaging Materials in Disposal Relevant Brines.” *Nuclear Technology*. Vol. 104. pp. 343–350.

Smailos, E. and R. Köster. 1987aa. “Corrosion Studies on Selected Packaging Materials for Disposal of High Level Wastes.” *Materials Reliability in the Back End of the Nuclear Fuel Cycle*. Proceedings of a Technical Committee Meeting, Vienna, Austria, September 2–5, 1986. TECHDOC–421. Vienna, Austria: International Atomic Energy Agency. pp. 7–24.

Smith, D. 2003aa. "Weld Flaw Evaluation and Nondestructive Examination Process Comparison Results for High-Level Radioactive Waste Package Manufacturing Program." TDR–EBS–ND–000007. Rev. 01. ACC: ENG.20030515.0003. Las Vegas, Nevada: Bechtel SAIC Company.

SNL. 2009aa. "Unexpected Test Results—Heterogeneous Alloy 22 Oxide Thickness." CR 12799. DEN001614731. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2009ab. "Unexpected Test Results—Residue on Subset of Alloy 22 Coupons." CR 12868. DEN001614752. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ac. "Features, Events, and Processes for the Total System Performance Assessment: Methods." ANL–WIS–MD–000026. Rev. 00. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aa. "Analysis of Mechanisms for Early Waste package/Drip Shield Failure." ANL–EBS–MD–000076. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aj. "EBS Radionuclide Transport Abstraction." ANL–WIS–PA–000001. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ak. "Engineered Barrier System: Physical and Chemical Environment." ANL–EBS–MD–000033. Rev. 06. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007al. "General Corrosion and Localized Corrosion of Waste package Outer Barrier." ANL–EBS–MD–000003. Rev. 03. ACN 01, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bb. "Stress Corrosion Cracking of Waste package Outer Barrier and Drip Shield Materials." ANL–EBS–MD–000005. Rev. 04. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2004ab. "Evaluation of Potential Impacts of Microbial Activity on Drift Chemistry." ANL–EBS–MD–000038. Rev. 01. ACN 02, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

Sprowls, D.O. 1987aa. "Evaluation of Stress-Corrosion Cracking." *Metals Handbook—Corrosion: Vol. 13*. 9th Edition. Metals Park, Ohio: ASM International. pp. 245–282.

Structural Integrity Associates. 2002aa. "Structural Integrity Associates Support of Waste package Design for Year 2001." LSN DEN001314737. San Jose, California: Structural Integrity Associates.

Thomas, D.E. and H.B. Bomberger. 1983aa. “The Effects of Chlorides and Fluorides on Titanium Alloys in Simulated Scrubber Environments.” *Materials Performance.* Houston, Texas: NACE International. pp 29–36.

U.S. Nuclear Waste Technical Review Board. 2001aa. “Proceedings From the International Workshop on Long-Term Extrapolation of Passive Behavior Conference.” Arlington, Virginia, July 19–20, 2001. A.A. Sagues and C.A. Di Bella, eds. Arlington, Virginia: U.S. Nuclear Waste Technical Review Board.

Wong, L., T. Lian, D.V. Fix, M. Sutton, and R. Rebak. 2004aa. “Surface Analysis of Alloy 22 Coupons Exposed for Five Years to Concentrated Ground Waters.” Proceedings of the CORROSION 2004 Conference. Paper No. 04701. Houston, Texas: NACE International.

Woolf, R. 2003aa. “Controlled Plasticity Burnishing (CPB) for Developing a Very Deep Layer of Compressive Residual Stresses in Rectangular Specimens of Alloy 22 for Yucca Mountain Nuclear Waste package Closure Weld.” SET Job No. 37. LSN DN2002365233. Cincinnati, Ohio: Surface Enhancement Technologies.

Yang, L., D. Dunn, G. Cragnolino, X. He, Y.-M. Pan, A. Csontos, and T. Ahn. 2007aa. “Corrosion Behavior of Alloy 22 in Concentrated Nitrate and Chloride Salt Environments at Elevated Temperatures.” Proceedings of the CORROSION/2007 Conference. Paper No. 077580. Houston, Texas: NACE International.

Yang, L., R. Pabalan, and M. Juckett. 2006aa. “Deliquescence Relative Humidity Measurements Using an Electrical Conductivity Method.” *Journal of Solution Chemistry*. Vol. 35, No. 4. pp. 583–604.

CHAPTER 5

2.2.1.3.2 Mechanical Disruption of Engineered Barriers

2.2.1.3.2.1 Introduction

This chapter of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the proposed Engineered Barrier System (EBS) the U.S. Department of Energy (DOE) described in its Safety Analysis Report (SAR), Section 2.3.4 (DOE, 2008ab). The design aspects of the EBS were described in SAR Sections 1.3.4 and 1.5.2, while the performance aspects were described in SAR Sections 2.1, 2.3.4, 2.3.5, 2.3.6, and 2.3.7. DOE stated that the following engineered barrier system features contribute to barrier performance: emplacement drifts, drip shields, waste packages, waste forms, waste form internals, waste package pallets, and emplacement drift inverts. According to DOE, the engineered barrier system features are designed to work together with the natural barriers by preventing or substantially reducing the release rate of radionuclides from the repository to the accessible environment. A disruption of the engineered barrier system components has the potential to affect their barrier performance. DOE anticipates that mechanical disruption of engineered barrier system components could generally result from external loads generated by accumulating rock rubble. Rubble accumulation can result from processes such as (i) degrading emplacement drifts due to thermal loads, (ii) time-dependent natural weakening of rocks, and (iii) effects of seismic events (vibratory ground motion or fault displacements). This TER chapter evaluates the performance of the various engineered barrier system components under a reasonable range of anticipated loading conditions.

To estimate the timing and extent of rubble accumulation, rocks in the repository block (RB) need geologic characterization. DOE characterized the repository rock mass as consisting of two major rock types: lithophysal and nonlithophysal. Lithophysal rocks (approximately 85 percent of the repository emplacement area) are characterized as relatively more deformable rocks with low compressive strength because of the voids of varying sizes contained within the rock. The nonlithophysal rocks (approximately 15 percent) are characterized as hard, strong, and jointed rocks. According to DOE, these two rock types are expected to behave differently under thermal and seismic loads and thus require different modeling approaches to account for different modes of failure (e.g., rock blocks separating from the mass and falling due to gravity, gradual unraveling or, tensile failure during vibratory loads). On the basis of geologic mapping and testing, DOE categorized the lithophysal rocks into five categories on the basis of rock mass qualities (to represent the variability in mechanical properties). DOE has conducted laboratory and *in-situ* testing on small and large rock samples and developed a range of input parameters for the numerical models. DOE has presented several approaches to estimate the timing and extent of degradation, including numerical modeling results.

According to DOE, the functions of the drip shield are to prevent rocks from falling on the waste packages and to prevent water from contacting the waste package surface soon after emplacement when waste packages are still hot, thereby minimizing the potential for corrosion. The purpose of the waste package is to protect the waste form and isolate the radionuclides or slow down their rate of release to the accessible environment. To estimate the effects on timing and magnitude of radionuclide release, DOE analyses considered potential loads from seismic events and the resulting mechanical disruption of the engineered barrier system components. DOE considered gradual drift degradation due to thermal loads, time-dependent weakening, and seismic events as sources of generating loads from rubble accumulation on and around the drip shields. However, DOE excluded the effects of drift degradation due to thermal loads and time-dependent weakening from its Total System Performance Assessment (TSPA) code. The NRC staff's review of DOE's technical bases for exclusion of features, events, and processes (FEPs) (FEP 2.1.07.02.0A, Drift Collapse) is presented in TER Section 2.2.1.2.1.3.2. The scope of this TER chapter is limited to reviewing how DOE considered the effects of seismic disruption (i.e., vibratory ground motion and fault displacement) and used the results in the Total System Performance Analysis.

The U.S. Nuclear Regulatory Commission (NRC) staff's review followed the guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). YMRP Section 2.2.1 provides guidance to the NRC staff on applying risk information throughout the review of the performance assessment. The NRC staff used DOE's risk information derived from an initial review of DOE's treatment of multiple barriers, as appropriate. The NRC staff's review approach is to assess the DOE design and analyses of engineered barrier system components under anticipated demands generated by drift degradation due to seismic loads. For those cases in which the design capacities may be exceeded, the NRC staff examined the potential for continued functionality of the components under a range of anticipated conditions. On the basis of the risk insights developed, the NRC staff's review focuses primarily on the seepage barrier functionality of the drip shield and the potential for loads from accumulated rubble to be transferred onto the waste package. In considering the range of possible loads and temperature conditions that can be generally anticipated during the repository life, the NRC staff takes into account uncertainty and variability in (i) rock characterization data, (ii) laboratory and *in-situ* test results, (iii) modeling approaches and conceptualization of failure modes, (iv) independent NRC staff's verifications, and (v) professional judgment based on experience and an understanding of excavated rock behavior under thermal and seismic loads.

2.2.1.3.2.2 Evaluation Criteria

NRC staff's assessment of model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for mechanical disruption of engineered barriers is guided by criteria in 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The DOE Total System Performance Assessment is reviewed in TER Section 2.2.1.4.1.

The regulations for performance assessment in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

- The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) sets forth requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 set forth requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, the DOE dose calculation is to

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure

- Include in the performance assessment those features, events, and processes used in the performance assessment for the initial 10,000-year period

This model abstraction of mechanical disruption of engineered barriers involves seismic and igneous activity. Thus, 10 CFR 63.342(a) and (b) also apply to this abstraction as these regulations require that DOE provide criteria for assessing the effects of seismic and igneous activity on the repository performance, subject to the probability limits in 10 CFR 63.342(a) and (b). Specific constraints on the analysis required for seismic and igneous activity analyses are given in 10 CFR 63.342(c)(1)(i) and 10 CFR 63.342(c)(1)(ii), respectively.

The NRC staff's review of the SAR and supporting information follows the guidance laid out in YMRP Sections 2.2.1.3.2, Mechanical Disruption of Engineered Barriers, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions provide guidance on information NRC staff could use to evaluate the performance assessment. Following the guidance, the NRC staff's review of DOE's abstraction of mechanical disruption of engineered barriers considered five criteria

1. System description and model integration are adequate.
2. Data are sufficient for model justification.
3. Data uncertainty is characterized and propagated through the abstraction.
4. Model uncertainty is characterized and propagated through the abstraction.
5. Model abstraction output is supported by objective comparisons.

10 CFR Part 63 specifies the use of a risk-informed approach for the review of a license application. The NRC staff review used the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of mechanical disruption of engineered barriers important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE and on NRC staff's knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.2.3 Technical Evaluation

2.2.1.3.2.3.1 Seismic and Fault Displacement Inputs for Mechanical Disruption of Engineered Barriers

DOE investigated the geological, geophysical, and seismic characteristics of the Yucca Mountain region to obtain sufficient information to estimate how the site would respond to vibratory ground motions from earthquakes. In SAR Section 1.1.5.2, DOE provided its description of site seismology. DOE described its analysis of potential seismic hazards in SAR Section 1.1.5.2.4, the overall approach to developing a seismic hazard analysis for Yucca Mountain in SAR Section 2.2.2.1, and the conditioning (adaption or modification) of the ground motion hazard at Yucca Mountain in SAR Section 1.1.5.2.5.1. Additional information was provided in DOE responses to the NRC staff's request for additional information (RAI) in DOE Enclosure 19 (2009ab) and DOE Enclosures 6, 7, and 8 (2009aq), and the references cited therein.

The DOE overall approach to developing a seismic hazard analysis for Yucca Mountain, including fault displacement hazards as described in SAR Section 2.2.2.1, involves the following three steps:

1. Conducting an expert elicitation in the late 1990s to develop a probabilistic seismic hazard analysis (PSHA) for Yucca Mountain. This assessment included a probabilistic fault displacement hazard analyses (PFDHA) (CRWMS M&O, 1998aa; BSC, 2004bp). The probabilistic seismic hazard analysis was developed for a reference bedrock outcrop, specified as a free-field site condition with a mean shear wave velocity (V_S) of 1,900 m/sec [6,233 ft/sec] and located adjacent to Yucca Mountain. This value was derived from a V_s profile of Yucca Mountain with the top 300 m [984 ft] of tuff and alluvium removed, as provided in Schneider, et al., Section 5 (1996aa).

2. Conditioning probabilistic seismic hazard analysis ground motion results to constrain the large low-probability ground motions to ground motion levels that, according to DOE, are more consistent with observed geologic and seismic conditions at Yucca Mountain, as provided in BSC ACN02 (2005aj).

3. Modifying the conditioned probabilistic seismic hazard analysis results using site-response modeling. This accounts for site-specific rock material properties of the tuff, in and beneath the emplacement drifts, and the site-specific rock and soil material properties of the strata beneath the site.

Probabilistic Seismic Hazard Analysis Methodology

DOE conducted an expert elicitation on probabilistic seismic hazard analysis in the late 1990s (CRWMS M&O, 1998aa; BSC, 2004bp) on the basis of the methodology described in the Yucca Mountain Site Characterization Project (DOE, 1997aa). DOE stated that its probabilistic seismic hazard analysis methodology followed the guidance of the DOE-NRC-Electric Power Research Institute-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa). In SAR Section 2.2.2.1.1.1, DOE concluded that the methodology used for the probabilistic seismic hazard analysis expert elicitation is consistent with NRC expert elicitation guidance, which is described in NUREG–1563 (NRC, 1996aa).

To conduct the probabilistic seismic hazard analysis, DOE convened two panels of experts as described in SAR Section 2.2.2.1.1.1. The first expert panel consisted of six 3-member teams of geologists and geophysicists (seismic source teams) that developed probabilistic distributions to characterize relevant potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates incorporated random uncertainties that were specific to the regional crustal conditions of the western Basin and Range. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the probabilistic seismic hazard analysis. The two expert panels were supported by technical teams from DOE, the U.S. Geological Survey, and Risk Engineering Inc., which provided the experts with relevant data and information; facilitated the formal elicitation, including a series of workshops designed to accomplish the elicitation process; and integrated the hazard results.

According to the DOE-NRC-Electric Power Research Institute-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa), the basic elements of the probabilistic seismic hazard analysis process are (i) identification of seismic sources such as active faults or seismic zones; (ii) characterization of each of the seismic sources in terms of their activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) ground motion attenuation relationships to model the distribution of ground motions that will be experienced at the site when a given magnitude earthquake occurs at a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, including associated uncertainties. According to the Budnitz, et al. (1997aa) methodology, each logic tree pathway represents one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that is representative of the seismic hazard at a site, including variability and uncertainty.

NRC Staff's Evaluation of DOE's Probabilistic Seismic Hazard Analysis Methodology

The NRC staff reviewed DOE's probabilistic seismic hazard analysis (PSHA) methodology described in SAR Sections 1.1.5.2.4 and 2.2.2.1.1 using the guidance provided in the YMRP and NUREG–1563 (NRC, 1996aa). The NRC staff also evaluated the DOE probabilistic seismic hazard analysis development to ensure that it included the four basic elements described in Budnitz, et al. (1997aa). In addition, the NRC staff observed all expert elicitation meetings and reviewed summary reports of those meetings as they were produced. On the basis of this information, including the evaluation with respect to Budnitz, et al. (1997aa) and NRC staff's direct observations of the expert elicitation process, the NRC staff notes that DOE's elicitation for the probabilistic seismic hazard analysis is consistent with the framework for conducting an expert elicitation described in NUREG–1563 (NRC, 1996aa), as referenced in the YMRP.

Probabilistic Seismic Hazard Analysis—Input Data and Interpretations

During the expert elicitation, DOE's seismic source teams considered a range of information from many resources including DOE, the U.S. Geological Survey, project-specific Yucca Mountain studies, and information published in the scientific literature. This information included (i) data and models for the geologic setting as summarized in BSC (2004bp); (ii) seismic sources and seismic source characterization, including earthquake recurrence and maximum magnitude (BSC, 2004bp); (iii) historical and instrumented seismicity, as described in CRWMS M&O Appendix G (1998ab); (iv) paleoseismic data (Keefer, et al., 2004aa); and (v) ground motion attenuation (Spudich, et al., 1999aa). DOE also supported the probabilistic seismic hazard analysis with a broad range of data, process models, empirical models, and seismological theory (CRWMS M&O, 1998ab). The expert panels built their respective inputs to the probabilistic seismic hazard analysis on the basis of this information and information presented to the experts during the elicitation meetings (CRWMS M&O, 1998ab). The resulting set of hazard curves was intended to provide DOE with sufficient representation of the seismic hazard for use in the TSPA analysis.

DOE expressed the probabilistic seismic hazard analysis (PSHA) curves in increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves were developed for bedrock conditions with a mean shear wave velocity (V_s) of 1,900 m/sec [6,233 ft/sec]. Such rocks are located adjacent to Yucca Mountain as described previously in the probabilistic seismic hazard analysis methodology subsection of this TER section. Estimates of uncertainty in the hazard curves are also included (see SAR Figure 1.1-74; e.g., hazard curves). The SAR provided probabilistic seismic hazard analysis results on horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations (SAs) at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and peak ground velocity (PGV).

NRC Staff's Evaluation of Input Data and Interpretations

The NRC staff reviewed DOE's probabilistic seismic hazard analysis input data and interpretations as described in SAR Sections 1.1.5.2 and 2.2.2.1.1. The NRC staff notes that DOE adequately developed geological, geophysical, and seismological information to support the expert elicitation. This conclusion is based in part on the NRC staff's evaluations of this information in NUREG–1762 (NRC, 2005aa). In NUREG–1762, the NRC staff concluded that the existing DOE information was consistent with site conditions at Yucca Mountain. This adequacy is also based on the NRC staff's first-hand knowledge of the geology and seismic characteristics of the Yucca Mountain region, which includes more than a decade of independent geological and geophysical research and study (e.g., Ferrill, et al., 1999ab, 1996aa; Stamatakos, et al., 2007aa, 1998aa,1997ab; Waiting, et al., 2003aa; Gray, et al., 2005aa). The resulting suite of ground motion hazard curves; horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and peak ground velocity is reasonable because it is consistent with NRC guidance outlined in Regulatory Guide 1.165 (NRC, 1997ab) and Regulatory Guide 3.73 (NRC, 2003ae).

The NRC staff also reviewed additional geological, geophysical, and seismological information discovered since the elicitation was performed (Wernicke, et al., 2004aa). On the basis of the NRC staff's detailed understanding of the Yucca Mountain geology, new geological and seismological information would not substantially alter the probabilistic seismic hazard analysis results.

Conditioning of Ground Motion Hazard

DOE provided in SAR Section 1.1.5.2.5.1 the conditioning of ground motion hazard at the reference bedrock outcrop where the probabilistic seismic hazard analysis was developed. Since completion of the probabilistic seismic hazard analysis in 1998, several studies and reports, including ones from NRC staff (NRC, 1999aa), the Nuclear Waste Technical Review

Board Panel on Natural System and Panel on Engineered Systems (Coraddini, 2003aa), and DOE itself (BSC, 2004bj), questioned whether the very large ground motions the probabilistic seismic hazard analysis predicted at low annual exceedance probabilities (below ~10^{-6}/yr) were physically realistic. These ground motion values are well beyond the limits of existing earthquake accelerations and velocities from even the largest recorded earthquakes worldwide. They are deemed physically unrealizable because they require a combination of earthquake stress drop, rock strain, and fault rupture propagation that cannot be sustained without wholesale fracturing of the bedrock (Kana, et al., 1991aa).

The NRC staff notes that overly conservative earthquake ground motions arise in the DOE study because the seismic hazard curves were constructed as unbounded lognormal distributions. In past practice, probabilistic seismic hazard curves were used to estimate ground motions with annual exceedance probability to 10^{-4} or 10^{-5} (typical annual exceedance probability values for nuclear power plant design and safe shutdown earthquakes).

For Yucca Mountain, however, the seismic hazard curves were extrapolated to estimate ground motions with annual exceedance probabilities as low as 10^{-8} (SAR Section 1.1.5.2.5.1). At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated Gaussian distributions (the tail is not defined by the data, but by the assumed distribution) of the input ground motion attenuation models (Bommer, et al., 2004aa). As Anderson and Brune (1999aa) pointed out, overestimates of the hazards may also arise because of the way in which uncertainty in ground motion attenuation from empirical observations or theory is distributed between aleatory and epistemic uncertainties.

To account for these large ground motions, DOE modified or conditioned the hazard using both a shear-strain-threshold approach and an extreme-stress-drop approach, as described in SAR Section 1.1.5.2.5.1. Rather than reconvene the probabilistic seismic hazard analysis expert elicitation and redo the probabilistic seismic hazard analysis, DOE chose to treat the issue as part of the ground response analysis. Accordingly, DOE's second step in developing ground motion inputs for postclosure analysis, after the development of the probabilistic seismic hazard analysis, is to condition the ground motion hazard. This second step in the three-step DOE process includes information on the level of extreme ground motion that is consistent with the geological setting of Yucca Mountain. Conditioning of ground motion hazard is a unique study developed for the Yucca Mountain project.

NRC Staff's Evaluation of Conditioning Methodology

The NRC staff reviewed DOE's methods for conditioning its probabilistic seismic hazard analysis results in SAR Section 1.1.5.2.5.1; reviewed DOE's responses to the NRC staff's RAIs, DOE Enclosures 6 and 8 (2009aq); and notes the methods are reasonable because they are based on the shear-strain-threshold and extreme-stress-drop approaches, both of which are technically sound methods based on fundamental principles of physics and are widely accepted.

Results of Probabilistic Seismic Hazard Analysis Conditioning

The unconditioned hazard curve DOE developed, which is the annual probability of exceedance (APE) as a function of ground motion, is convolved with the distribution of extreme ground motion for the reference bedrock outcrop to produce the conditioned ground motion hazard of the same bedrock outcrop. The shear-strain-threshold conditioning has a marginal impact as compared to the extreme-stress-drop approach. For example, for an annual probability of exceedance of 10^{-8}, the shear-strain-threshold conditioned peak ground velocity hazard is reduced from 1,200 cm/sec to about 1,100 cm/sec [472 to 433 in/sec] or about 10 percent; the Stress-drop-conditioned peak ground velocity hazard is reduced from 1,200 cm/sec to about 480 cm/sec [472 to 189 in/sec] or about 60 percent, as identified in BSC Section A4.5.1 (2008bl). The impact of conditioning at low probabilities is less significant and increases as the probability decreases (i.e., annual probabilities of exceedance of 10^{-5}, 10^{-6}, 10^{-7}, and 10^{-8}) (SAR Section 1.1.5.2.5.1). SAR Figures 1.1-79 and 1.1-80 compared the unconditioned and conditioned peak ground accelerations (PGAs) and peak ground velocity (PGV) mean hazard curves for the reference bedrock outcrop.

BSC Appendix A (2008bl) outlined the workshop proceedings, which included presentations, discussions, and assessments, that were conducted to develop the expert judgment. The stress drop data from the United States and other countries were used in the expert judgment. The parameter variability involved in the empirical ground motion attenuation relationship and numerical simulations of ground motions that the experts relied on was included in the conditioning. Variability in velocity profile, stress drop, source depth, and kappa (the site- and distance-dependent parameter representing the effect of intrinsic attenuation of the wave field as it propagates through the crust from source to the receiver) were considered in the modeling to map the stress drop into ground motion distribution.

In response to NRC RAIs (DOE, 2009aq), DOE provided information showing DOE’s application of the two methods in series where the output of the extreme-stress-drop conditioning becomes the input of the shear-strain-threshold conditioning. In the RAI responses, DOE also clarified and updated the formulations for the two conditioning methods, as described in BSC Appendix A (2008bl).

NRC Staff’s Evaluation of Results of Conditioning

The conditioned hazard curves for postclosure Total System Performance Assessment (TSPA) calculations are appropriate because the conditioning methods follow basic mechanical, material, and seismological principles in which (i) the shear-strain-threshold approach is based on laboratory rock mechanics data and corroborated by numerical modeling and (ii) the extreme-stress-drop approach is based on the worldwide observations of stress drop from large earthquakes (SAR Section 1.1.5.2.5.1). The final conditioned ground motion levels at very low annual probability of exceedance (APE) continue to be very conservative when compared with the observed worldwide strong motion data, which include records from earthquakes much greater than those expected in the Yucca Mountain region. DOE assumed that the tectonic setting and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly during the next million years. This assumption is reasonable given the basic tectonics in the Yucca Mountain region and provides the basis for the conditioning at very low annual probability of exceedance.

Summary of NRC Staff’s Evaluation of Probabilistic Seismic Hazard Analysis

The NRC staff reviewed SAR Sections 1.1.5.2 and 2.2.2.1 and DOE’s responses to the NRC staff’s request for additional information (RAIs) and notes that the methodologies, input data, interpretation of the probabilistic seismic hazard analysis, and conditioning of the probabilistic seismic hazard analysis results for the Yucca Mountain site are reasonable. DOE used appropriate methods for relying on the collective judgment of appropriate experts by following an established procedure to elicit and document the experts’ conclusions. DOE supported the expert elicitation program with sufficient technical and scientific information.

2.2.1.3.2.3.1.1 Seismic Site-Response Modeling

To address the effects of earthquakes during the postclosure period, DOE performed site-response analysis, which incorporates the effects of the upper rock and soil layers on the input ground motion at the reference rock (the conditioned ground motion hazards discussed previously).

Overall Approach to Site-Response Modeling

In SAR Section 1.1.5.2.5.2 DOE discussed how the various types and thicknesses of rocks, alluvium, and soils that comprise the site would likely respond to earthquake ground motions. The results of site-response modeling include understanding and quantifying the amplification or damping factor of ground motion and how much the vertical-to-horizontal motion ratio varies from place to place. DOE used the site-specific ground motion curves that are consistent with the conditioned probabilistic seismic hazard analysis ground motion hazard curves.

NRC Staff's Evaluation of Site Response Modeling Approach

The NRC staff reviewed DOE's overall approach to site-response modeling using the guidance of NUREG/CR–6728 (McGuire, et al., 2001aa) and the YMRP. DOE's adoption of Approach 3 from NUREG/CR–6728 is a reasonable approach and is recommended by NUREG/CR–6728. The NRC staff also notes that the two frequency ranges (1–2 and 5–10 Hz) used in the calculations of input control motions are consistent with NRC guidance provided in NRC Regulatory Guide 1.165, Appendix C (NRC, 1997ab).

On the basis of the NRC staff's review described previously, DOE's method used to perform seismic site-response modeling, which is based on the random vibration theory, is reasonable. The NRC staff notes that the random-vibration-theory-based point-source and one-dimensional equivalent-linear site-response model combined with Approach 3 from NUREG/CR–6728 and the conditioned hazard as input are appropriate for use in characterizing the seismic ground motions at Yucca Mountain, because it accounts for the site response adequately without underpredicting the response. DOE's site-response model is reasonable for use in the performance assessment.

Ground Motion Inputs

DOE provided ground motion inputs developed for the repository block in SAR Section 1.1.5.2.6. For the site surface, 52 combinations of site properties were evaluated in the site-response modeling (SAR Section 1.1.5.2.6.1). These combinations were from two base case velocity profiles (south and northeast of the Exile Hill Fault splay), two base case sets of dynamic material property curves for tuff and alluvium separately, four values of alluvium thickness northeast of the fault splay, and three values of alluvium thickness south of the fault splay (resulting in a total of seven combinations). Each combination incorporated aleatory variability by averaging the amplification factors from 60 randomized velocity profiles and dynamic material property curves.

Site-Specific Hazard Curves

The seven combinations of alluvium and tuff hazard curves were combined into two sets: the northeast and south fault splay sets. The four and three combinations of hazard curves for four and three alluvium thicknesses were enveloped separately for south and northeast of the fault splay. These two sets of hazard curves were enveloped again to produce mean horizontal and vertical hazard curves (BSC, 2008bl). The final mean horizontal and vertical hazard curves for peak ground accelerations (PGAs); 0.05, 0.1, 0.2, 0.5, 1.0, 2.0, and 3.3 seconds spectral acceleration (SA); and peak ground velocity were provided in BSC Figures 6.5.2-34 to 6.5.2-42 (2008bl) for the surface facilities area and BSC Figures 6.5.3-9 to 6.5.3-16 (2008bl) for the repository block. The data for these plots are in DTN: MO0801HCUHSSFA.001, as identified in BSC Section 6.5.2.2 (2008bl).

Earthquake Time Histories

The repository block time histories for postclosure analyses were developed differently for annual probability of exceedance (APE) of 10^{-5}, 10^{-6}, and 10^{-7} (SAR Section 1.1.5.2.6.2), where 17 sets of time histories were developed: one horizontal (H1) component of each seed time history was scaled according to the peak ground velocity from site-response modeling and the other two components were scaled to maintain the inter-component variability of the seed time history (SAR Section 1.1.5.2.6.2).

NRC Staff's Evaluation of Ground Motion Inputs

The processes and procedures DOE used to develop site-specific hazard curves, time histories, strain-compatible soil properties for the site, and the ground motions for postclosure analyses are reasonable. DOE used an averaging process to account for the data (velocity profiles and dynamic material properties) and site-response model uncertainties and enveloping process to accommodate the alluvium thickness change (spatial variability) when it developed the hazard curves. Then DOE followed the recommended (McGuire, et al., 2001aa) routine procedures in engineering seismology for ground motion inputs. The strain-compatible soil properties are the products of the previously described site-response analysis.

2.2.1.3.2.3.1.2 Fault Displacement Hazard Analysis

Fault displacement (the relative displacement between opposite sides of a fault) is a potential hazard to the underground facility because it could damage or shear drifts and/or waste packages, trigger rockfall within the drifts and shafts, degrade drift walls and ground-support systems, and degrade other engineered barrier system components. These hazards might affect the postclosure performance of the engineered barriers.

Probabilistic Fault Displacement Hazard Analyses—Input Data and Interpretations

The probabilistic fault displacement hazard analyses integrated two data types: (i) known and/or documented faulting activity consisting of measurements of regional and local earthquakes, and measurements of fault displacements within the last ~1.8 million years and (ii) inferred potential faulting activity, on the basis of analysis of mapped geological faults, overall tectonic setting, and regional estimates of ongoing crustal strain. DOE analyzed 100 earthquakes in the Basin and Range region to determine the relationships among the amounts and patterns of both principal and distributed fault displacements, the minimum magnitude at which an earthquake may produce surface faulting, and the maximum magnitude at which an earthquake does not displace the surface.

For the largest mapped faults at Yucca Mountain, the probabilistic fault displacement hazard curves were largely based on the same detailed paleoseismic and earthquake data used to characterize these faults as potential seismic sources. The expert elicitation relied on both anecdotal evidence and expert judgment to develop conceptual models of distributed faulting and to estimate the probabilities of secondary faulting of smaller faults and fractures in the repository (Youngs, et al., 2003aa; CRWMS M&O, 1998aa).

DOE chose the following nine sites around Yucca Mountain as demonstration sites of the application of the probabilistic fault displacement hazard analyses, as identified in SAR Chapter 1, Table 1.1-67, p. 1.1-304: (i) Bow Ridge fault, (ii) Solitario Canyon fault, (iii) Drill Hole Wash fault, (iv) Ghost Dance fault, (v) Sundance fault, (vi) an unnamed fault west of Dune Wash, (vii) a location 100 m [328 ft] east of the Solitario Canyon fault, (viii) a location between Solitario Canyon fault and Ghost Dance fault, and (ix) a location within Midway Valley. These demonstration sites were selected to represent a range of faulting and related fault deformation conditions at the site, including large block-bounding faults, such as the Solitario Canyon and Bow Ridge faults; smaller mapped faults within the repository footprint, such as the Ghost Dance fault; and unmapped minor faults near the larger faults, fractured tuff, and intact tuff.

Results of the probabilistic fault displacement hazard analyses (CRWMS M&O, 1998aa) show that, except for the Bow Ridge and Solitario Canyon faults, mean fault displacements are less than 1 m [3.28 ft] over the next 10 million years (SAR Table 2.2-15). Mean displacements for the demonstration sites within the current repository footprint [demonstration sites (v), (vii), and (viii) as identified in the previous paragraph] do not exceed 0.40 m [1.3 ft] in 10 million years. For a 10,000-year period, the mean displacements are calculated to be less than 0.01 m [0.03 ft] for all 9 demonstration sites (SAR Table 1.1-67).

Individual fault displacement hazard curves were developed to characterize fault displacements at each of the nine demonstration sites. These fault displacement hazard curves are analogous to seismic hazard curves, in which increasing levels of fault displacements are computed as a function of the annual probability that those displacements will be exceeded. Example fault displacement curves for the nine demonstration sites were provided in SAR Figure 2.2-13.

NRC Staff's Evaluation of DOE's Probabilistic Fault Displacement Hazard Analyses

The NRC staff evaluated DOE's input to the probabilistic fault displacement hazard analyses in the SAR and supporting documents. The NRC staff also conducted its independent analysis of slip tendency (Morris, et al., 1996aa) and faults within the Yucca Mountain region (Morris, et al., 2004aa). The input data to the probabilistic fault displacement hazard analyses and their interpretation are appropriate because (i) the DOE approach used representative data, (ii) the methods used to interpret the data were rigorous, and (iii) the interpretations made are consistent with the regional characteristics. On the basis of the NRC staff's review, as described previously, DOE's methodology, input data, and interpretations of the probabilistic fault displacement hazard analyses are reasonable for use in the performance assessment.

2.2.1.3.2.3.2 Fault Displacement Considerations in TSPA

This section reviews the information provided in SAR Section 2.3.4.3 (and selected references) to evaluate the adequacy of the conceptual model of seismic fault displacement. DOE considered seismic fault displacement as one of the modeling cases in the TSPA. DOE assumed that fault displacement occurred concurrently with the ground motion during a low probability seismic event (SAR Section 2.3.4.5.5.1). DOE considered that only the waste packages located directly above faults were subject to damage from fault displacement. DOE expected the dose related to fault displacement to be a small fraction of the total dose for the seismic scenario class because damage from fault displacement affected a small fraction of the engineered barrier system and damage occurred only for events with very low exceedance frequencies. DOE calculated the dose contribution from the seismic fault displacement on the basis of simplified calculations (SAR Section 2.4.2.2.1.2.2.2). The abstraction assumed waste package damage when fault displacement exceeded the available waste package clearance. To evaluate fault displacement, DOE assumed (i) the faults are perpendicular to the drift axis with the displacement being vertical; (ii) the fault displacement occurs at a discrete plane, creating a sharp discontinuity; and (iii) clearances are based on emplacement drifts that are fully collapsed at the time of the seismic event (SAR Section 2.3.4.5.5.2.1.1).

The potentially relevant features, events, and processes in DOE's TSPA model were listed in SAR Table 2.2-1. In this abstraction, DOE evaluated and included FEP 1.2.02.03.0A, Fault Displacement Damages Engineered Barrier System Components.

Conceptual Model

Clearance

DOE analyzed the clearances between the engineered barrier system components for intact and failed drip shield scenarios (SAR Section 2.3.4.5.5.2.1). For intact drip shield configurations, DOE defined the clearance as the interior height of the drip shield less the outside diameter of the waste package outer corrosion barrier without a pallet, to account for part of the substantial movement of the rubble (SAR Section 2.3.4.5.5.2.1.1). On the basis of this simplification, DOE used a maximum allowable displacement with drift collapse and an intact drip shield that varied from 67.3 to 96.9 cm [26.5 to 38.15 in] (SAR Table 2.3.4-52), according to the type of waste package.

NRC Staff's Evaluation of Clearance

The NRC staff compared the values of DOE's clearances in SAR Table 2.3.4-52 to the engineered barrier system geometry presented in SAR Figure 2.3.4-53 that showed the distances between the top of the waste packages and bottom of the drip shields {35.6 to 68.6 cm [14 to 27 in]}. Additionally, the NRC staff considered the potential for the rubble to accommodate some of the fault displacement through compaction. On the basis of a confirmatory calculation, the NRC staff estimates that approximately 72–186 cm [28.3–73.2 in] of fault displacement can be accommodated through rubble compaction and the distance between the top of the waste package and the bottom of the drip shield crown. The values DOE used are adequate for their intended use because they fall within the range of potential displacements, on the basis of the design geometries.

For failed drip shield configurations, DOE stated that no free space existed between the top of the waste package and bottom of the drip shield. DOE concluded that rubble movement will accommodate some amount of fault displacement due to rubble consolidation (SAR Section 2.3.4.5.5.2.1.2). DOE stated that fault displacement had to exceed one-quarter of the outer diameter of the outer corrosion barrier to cause waste package failure {from 43.7 to 51.1 cm [17.2 to 20 in]} (SAR Tables 2.3.4-50 and 2.3.4-53). This approach is reasonable on the basis of a simplified, bounding calculation DOE presented in SNL Section 6.11.1.2 (2007ay). DOE showed that the potential porosity values for the rubble, if compacted, would accommodate 0.5–1.6 m [1.64–5.25 ft] of displacement. Accordingly, the low estimate of porosity, if compacted, generally accommodates the clearance DOE estimated.

Expected Movements and Number of Impacted Waste Packages

DOE stated that the probability of events exceeding 0.1 cm [0.039 in] of displacement in the repository block was 10^{-5} per year (SAR Section 2.3.4.5.5.1). DOE characterized the subsurface geologic repository operations area and determined that few faults were capable of producing movements greater than calculated clearances for events with a probability of exceedance of 10^{-8} per year (SAR Table 2.3.4-55). SAR Table 2.3.4-59 showed that less than 2 percent of the waste packages can potentially be impacted by a seismic faulting event with an annual exceedance frequency of 1×10^{-8}/yr to 3×10^{-8}/yr. To mitigate the potential risk of faulting that could cause mechanical damage to the waste packages, DOE stated that waste packages would be placed 60 m [196.85 ft] from known, major faults (SAR Table 1.9.9, Design Control Parameter 01-05). The NRC staff compared the probability, location, and magnitude of potential seismic fault displacement events determined in the probabilistic seismic hazard analysis (reviewed in TER Section 2.2.1.3.2.3) to the values presented for use in the TSPA (SAR Table 2.3.4-55) model and noted the values are consistent.

NRC Staff's Evaluation of Number of Impacted Waste Packages

On the basis of the low probability of occurrence, the limited number of faults that could impact the waste package via fault displacement, and the 60-m [196.85-ft] offset from the location of known faults capable of impacting the waste packages, DOE's estimate of the number of waste packages that could be potentially impacted by seismic faulting events is reasonable.

Damage to the Engineered Barrier System

DOE sampled a uniform distribution of open areas of waste packages to model the open area of a waste package failed by fault displacement. This distribution has a lower bound of 0 m^2 [0 ft^2] and an upper bound equal to the area of the waste package lid (SAR Section 2.3.4.5.5.4). DOE stated that the areas of the lids for the transportation, aging, and disposal (TAD) and codisposal (CDSP) waste package groups are 2.78 and 3.28 m^2 [30 and 35 ft^2], respectively. (These areas were calculated using the waste package diameters provided in SAR Table 2.3.4-50.)

NRC Staff's Evaluation of Engineered Barrier System Damage

The NRC staff notes the breached area distribution appropriately bounds the potential breach area expected in the fault displacement modeling case based on the following considerations. Given the uncertainty in the magnitude of displacement, the modeled breach area bounds the potential breach area, which could range from a plastically deformed waste package (with no open area) to one that has been completely sheared or sliced in half. Additionally, the average waste package is deemed to provide no barrier capability to crown seepage once approximately 4 percent {~1.5 m^2 [~16 ft^2]} of the surface area is breached. Thus, the waste package barrier capability to prevent water from contacting the waste form decreases proportionally. (This aspect of DOE's performance assessment is evaluated in TER Section 2.2.1.3.3.3.3.)

To determine the impact of seismic faulting on drip shield failure, DOE assumed that the drip shield fails at the instant the underlying waste package is breached. This simplification is reasonable because the magnitude of a faulting event required to damage a waste package would damage the adjacent drip shield. The NRC staff notes the loss of drip shield barrier capability is appropriately bounding, in that the drip shield retains no barrier capability.

Summary of NRC Staff's Evaluation of Fault Displacement

DOE adequately represented the fault displacement modeling case in the Total System Performance Assessment analysis because

- The clearances are reasonably bounding, when considering the potential for rubble compaction

- The expected fault displacements are consistent with the site characterization data

- The impact on the engineered barrier system appropriately bounds the potential consequences

2.2.1.3.2.3.3 Seismically Induced Drift Degradation

The NRC staff's evaluation in this TER section focuses on DOE's assessment of potential drift degradation due to seismic ground motions and use of the information to assess potential mechanical disruption of the engineered barriers. The NRC staff reviewed DOE's information in SAR Section 2.3.4.4 and supporting documents (BSC, 2004al) that describe potential seismically induced degradation of emplacement drifts after permanent closure. DOE's information included estimates of the amount of rubble accumulation in drifts, drip shield loading due to rubble, sizes of individual blocks that may strike the drip shield during a seismic event, and the associated impact velocity and location of the impact on the drip shield. The NRC staff's evaluation focuses on the potential occurrence of rubble loading that is large enough to damage the drip shield and the time of occurrence of such rubble loading. The time of occurrence is important because the NRC staff's review (TER Section 2.2.1.3.1) suggests that mechanical damage of the drip shield during approximately the first 12,000 years after repository closure could expose the waste package to aggressive chemical conditions that may support localized corrosion. DOE estimated the potential for rubble accumulation in drifts through process-level analyses of the effects of seismic ground motions on drift degradation. DOE chose the analysis approach by considering the mechanical behavior of two types of rocks (i.e., lithophysal and nonlithophysal) that constitute the repository block (SAR Section 2.3.4.4.8.3). The NRC staff's review of DOE's rock characterization information relevant to drift degradation modeling is summarized in TER Section 2.2.1.2.1.3.2.

DOE's Process-Level Modeling of Drift Degradation Due to Seismic Events

According to DOE, the mechanical deformation of the nonlithophysal rock mass will be controlled predominantly by movements of rock blocks along existing fractures. DOE analyzed drift degradation in nonlithophysal rock by modeling motions of rock blocks on surfaces formed by existing fractures as described in SAR Section 2.3.4.4.4. DOE used the analyses to estimate the characteristics of individual rock blocks that may strike the drip shields during a seismic event (SAR Table 2.3.4-19) and the volumes of rubble that may accumulate in the drifts (SAR Tables 2.3.4-20 and 2.3.4-24). However, the NRC staff's review focuses on potential rubble accumulation in the lithophysal rocks, which cover about 85 percent of the emplacement drifts. The potential for rock block impacts exists only in nonlithophysal areas (approximately 15 percent of the emplacement drifts). DOE considered the failure mechanism by rock block impact and excluded it from the TSPA model (excluded FEP 1.2.03.02.0B as a result of low consequence).

For the lithophysal rock mass, DOE indicated that mechanical deformation of the rock will consist predominantly of rock material deformations aided by lithophysae and a high density of existing small-scale fractures. As described in SAR Section 2.3.4.4.5, DOE analyzed drift degradation in lithophysal rock by modeling potential fracturing of the rock through formation of new fractures and movement on existing fractures as dictated by the rock stress. DOE used the analyses to estimate potential rubble accumulation for drift sections in lithophysal rock but applied the results of the analyses to the entire repository. DOE stated that this approach would be bounding.

To assess potential drift degradation in lithophysal rock, DOE used a model that focuses on estimating the rock mass volume that could break up along failure surfaces determined by the effects of the prevailing stress field on a diffuse network of small "incipient" fractures. DOE obtained the estimates by modeling the rock mass as an assemblage of polygonal blocks generated randomly using the Voronoi tessellation model, as identified in SAR Section 2.3.4.4.5.3 and BSC Sections 6.4.2.1 and 7.6.1 (2004al). The individual blocks can deform elastically or slide or separate at block contact surfaces. The blocks are initially attached together at contact surfaces and may slide or separate if the contact resistance is overcome by the prevailing stress. Thus, the contacts represent incipient fractures that could allow the blocks to detach from the assemblage if the prevailing stress permitted such detachment. Detached blocks could fall as a result of gravity or seismically induced force. DOE stated in BSC Section 6.4.2.1 (2004al) that the model could simulate rock deformation, stress changes, rock fracturing or breakage, and free fall of broken rock blocks. DOE explained that mechanical behavior of the model is influenced by the block size, block elastic parameters (Young's modulus and Poisson's ratio), contact elastic parameters (shear and normal stiffness), and contact strength parameters (tensile strength, cohesion, and friction). DOE set the values of the model parameters by calibrating the unconfined compressive strength and elastic stiffness of the model (i.e., block assemblage) against the unconfined compressive strength and elastic stiffness of the rock mass on the basis of laboratory test data, as outlined in BSC Section 7.6.1 (2004al). DOE implemented the model in a two-dimensional universal distinct element code UDEC, as identified in BSC Section 3.1 (2004al), and used the code to calculate changes in drift profile and amount of rubble accumulation due to seismic events, and drip shield loading due to rubble. Additional details of the rock characterization, laboratory and field testing, numerical experiments for calibration, and field validation are summarized and reviewed in TER Section 2.2.1.2.1.3.2. The application of this model for estimating the amount of rubble accumulation and the corresponding loads that might act on the drip shields and other components of the engineered barrier system is addressed in the following sections.

DOE's Model for Estimating Rubble Formation Due to Seismic Events

Consideration of the Two Rock Types in the TSPA Model

DOE stated that the rockfall volume in the nonlithophysal zones is significantly less than in the lithophysal zones for the same peak ground velocity level because the nonlithophysal rock mass is significantly stronger than the lithophysal rock. This assessment is reasonable on the basis of a comparison of strength and mechanical properties of the two rock types provided in SAR Tables 2.3.4-16 and 2.3.4-17. According to DOE's data (SAR Table 2.3.4-25), the mean rockfall volume in the lithophysal rock is a factor of 40 to 200 greater than the mean rockfall volume in the nonlithophysal rock for the 1.05 and 2.44 m/sec [3.44 and 8 ft/sec] peak ground velocity levels, respectively. On the basis of a review of DOE's information, the NRC staff notes that the weaker lithophysal rock will usually fail before nonlithophysal rock during a seismic event; thus, the probability of rockfall for the lithophysal rock defines the upper bound for the probability of rockfall in the nonlithophysal zones. DOE's use of 30–120 m^3/m [320–1,280 ft^3/ft] in the TSPA is reasonable for both rock types. The NRC staff verified that the rockfall volumes used for calculating the fraction of drift filled with rubble as a function of time presented in SAR Figure 2.1-14 are supported by model predictions of rubble volume calculated for lithophysal and nonlithophysal rocks (SAR Tables 2.3.4-23 and 2.3.4-24). Thus, DOE's approach of using a single bounding range to represent rockfall in both rock types does not underestimate the volume of rockfall in the nonlithophysal region.

Drift Degradation From Seismic Events

DOE indicated in SAR Section 2.3.4.4.5.4 that seismic ground motions could cause partial or complete collapse of drifts in lithophysal rock, resulting in various amounts of rubble accumulation for different ground motion magnitudes and mechanical categories of lithophysal rock (BSC, 2004al). To describe the effects of ground motion magnitude and lithophysal rock categories on potential rubble accumulation, DOE performed analyses for ground motion at peak ground velocity levels of 0.4, 1.05, and 2.44 m/s [1.31, 3.44, and 8 ft/sec], which, according to DOE, correspond to an annual frequency of exceedance of 10^{-4}, 10^{-5}, and 10^{-6}, respectively (SNL, 2007ay). DOE performed the analyses using 15 ground motion time history cases at each annual frequency of exceedance and 5 sets of values of mechanical properties representing the 5 lithophysal rock categories DOE defined in SAR Section 2.3.4.4.5.4 and BSC Section 6.4.2.2.2 (2004al). DOE concluded that (i) ground motion with an annual frequency of exceedance of 10^{-4} will have negligible effects on drift degradation, (ii) ground motion with an annual frequency of exceedance of 10^{-6} will cause complete drift collapse, and (iii) ground motion with an annual frequency of exceedance of 10^{-5} could cause various degrees of collapse and varying amounts of rubble accumulation. DOE selected 15 analysis cases (SAR Table 2.3.4-23) to represent potential rubble accumulation due to seismic ground motions, as described in SNL Section 6.7.1.1 (2007ay).

DOE used the calculated rubble volumes from 11 of the 15 cases to develop relationships between rubble accumulation and the peak ground velocity of a seismic event, as described in SAR Figure 2.3.4-48 and SNL Section 6.7.1.2, Figure 6-57 (2007ay). SAR Section 2.3.4.4.8.3.1 stated that four cases calculated using rock mass Category 1 properties were eliminated because DOE considered the rubble volumes from the four cases to be nonrepresentative. According to SAR Section 2.3.4.4.8.4, DOE used the resulting relationship between rubble volumes and peak ground velocity to estimate the amount of rubble accumulation due to a seismic event. DOE estimated the rubble accumulation due to multiple seismic events by adding the accumulations from the individual events, as identified in SAR Section 2.3.4.4.8.4 and SNL Section 6.7.1.4 (2007ay). The abstraction did not include weakening of the host rock due to previous events, because rapid filling of drifts in lithophysal units mitigates concerns about numerous seismic events slowly weakening the rock mass, as DOE explained in SNL Section 6.7.1.4 (2007ay). Therefore, to calculate the drift volume fraction filled with rubble in lithophysal rock areas after a sampled seismic event, DOE accumulated rockfall [using relationships based on SNL Table 6-30 and Figure 6-56 (2007ay)] and divided the accumulated volume by a number sampled from a uniform distribution between 30 and 120 m^3/m [320 and 1,280 ft^3/ft]. The uniform distribution of 30–120 m^3/m [320–1,280 ft^3/ft] represents DOE's estimate of the volume of rockfall that would fill a drift with rubble, as described in SNL Section 6.12.2 (2007ay). SAR Figure 2.1-14 summarized DOE's estimates of potential rubble accumulation due to seismic events during a 1-million year period. According to this figure, at 10,000 years, the volume of accumulated rubble in the drift could reach as much as 15 percent of the drift volume (with a mean value of approximately 4 percent). Similarly, the corresponding volume of accumulated rock rubble at 100,000 years reached about 60 percent (with a mean value of about 32 percent). The volume of rubble per unit length of drift that is required to fill the drift was sampled for each epistemic realization in the TSPA analysis, and the sampled volume ranged uniformly between 30–120 m^3/m [320–1,280 ft^3/ft].

To calculate the static load on a drip shield due to rubble, DOE multiplied the volume fraction of the drift filled with rubble (on the basis of the assessment of drifts in lithophysal rock) by the drip shield load of a fully collapsed drift, as outlined in SNL Section 6.12.2 (2007ay). The magnitude of drip shield loading due to rubble at a given time depends on the amount of rubble accumulation, shape of the rubble pile, and the amount of rubble loading transmitted to the drip shield for a given amount and shape of the rubble pile. DOE used the information to determine the potential occurrence of rubble loading large enough to damage the drip shield in lithophysal and nonlithophysal areas.

NRC Staff's Evaluation

The NRC staff evaluated DOE's assessment of potential drip shield loading due to rubble accumulation in lithophysal rock sections of emplacement drifts by considering the amount of rock accumulation and the shape of rubble piles.

Amount of Rubble Accumulation

The timing and amount of rubble accumulation in a drift due to seismic events depend on the occurrence of seismic ground motions large enough to cause rock failure around the drift opening. The occurrence of rock failure during a seismic event depends on the ground motion levels and the rock mass strength. The rock mass strength around an emplacement drift during a seismic event may be affected by any previous weakening of the rock due to thermal stress, time-dependent effects, or seismic ground motions. In SNL Section 6.7.1.4 (2007ay), DOE's abstraction of rubble accumulation due to seismic events did not include any effects of rock weakening, because, according to DOE, rapid filling of drifts in lithophysal units mitigates concerns about the effects of rock weakening. In response to an NRC staff's request for additional information, DOE provided information in DOE Enclosure 1 (2010aa) that showed the following: (i) seismic ground motions with an annual frequency of exceedance of 10^{-4} will have negligible effects on drift degradation if the host rock has not been affected by thermal stress; (ii) ground motions with an annual frequency of exceedance of 10^{-4} that occur in the presence of thermal stress in the rock could cause some additional drift damage and rubble accumulation, as outlined in DOE Enclosure 1, Figure 3(a) (2010aa); and (iii) the amount of drift damage and rubble accumulation due to the seismic event increase if the event occurs after a long period of time-dependent weakening of the rock, as described in DOE Enclosure 1, Figure 3(b)(2010aa).

DOE also showed, with appropriate technical basis, that the potential incremental rockfall due to (i) multiple events and (ii) combined effects of time-dependent weakening and seismic events was relatively small during the first 10,000 years, as detailed in DOE Enclosure 1, Figure 4 (2010aa). The NRC staff obtained additional confidence in DOE's conclusion as explained next, by comparing the analytical results to empirical mining industry data provided in DOE's response to the NRC staff's request for additional information, as identified in DOE Enclosure 1, Figure 5, (2010aa). DOE Enclosure 1, Figure 5 (2010aa) and BSC Figure 6-149 (2004al) summarized the caving potential of an excavated underground opening. Caving potential is expressed in terms of modified rock mass rating (a measure of rock quality and strength where a higher value indicates greater stability) and hydraulic radius [a dimension based on the geometry of the excavated opening; hydraulic radius is a measure of stability (i.e., a larger hydraulic radius indicates a less stable opening, and hence a higher caving potential)]. The rock mass rating data for Yucca Mountain rocks (between 50 and 60) show the corresponding hydraulic radii needed to cause high caving potential, which is on the order of 25 to 35 m [82 to 115 ft]. DOE analyses showed that the hydraulic radius of a degraded waste emplacement drift after 10,000 years of heating and time-dependent strength degradation coupled with a seismic ground motion would still be far less than the hydraulic radius of a degraded opening with high caving potential. Therefore, the following DOE conclusion is reasonable: while the incremental rockfall accumulation due to combined effects and cumulative effects could be considerable over the entire period of repository performance (hundreds of thousands of years), DOE in its TSPA analyses did not underestimate the amount of rubble accumulation for the initial 10,000 years of repository performance.

Shape of Rubble Piles

The process-level model that DOE used to analyze drift degradation in lithophysal rock appears somewhat constrained because of the upper boundary. In BSC Figure 6-116 (2004al), the tessellated domain (a mosaic pattern used to represent the domain with discrete elements) of the model is set 10.25 m [33.6 ft] above the initial drift roof. The upper boundary of the potential degradation zone DOE used was 1.86 drift diameters above the initial drift roof. However, contours of block displacement magnitude intersected the upper boundary of the tessellated domain. The calculated displacement contours indicate that some additional displacement could occur outside this domain [e.g., BSC Figure 6-176 (2004al)]. Also, plots of the final position of the Voronoi blocks after an analysis [e.g., BSC Figures P-17, P-18, and P-24 (2004al)] indicate blocks at the top of the model could be predicted to separate from the overlying elastic domain. Such a separation would suggest the caved zone might have extended higher if the model upper boundary had been higher.

In response to an NRC staff's request for additional information, DOE provided analyses to show that the rubble volume calculated using DOE's model is insensitive to the size of the tessellated domain (DOE, 2010aa). DOE provided results from two models with the boundary of the tessellated domain at 8.25 and 13.25 m [27 and 43.5 ft], respectively, in DOE Enclosure 2, Section 1.3 (2010aa). DOE used the models to estimate the extent of caving needed to fill a drift with rubble if the prevailing mechanical conditions were to cause complete drift collapse. This was accomplished by artificially degrading the rock mass strength to zero in a quasi-static analysis. The results of these studies demonstrated that the patterns of calculated displacements and stresses did not change appreciably as a result of changing the size of the tessellated domain. The quasi-static analyses showed that the caved zone extended approximately one drift diameter above the drift irrespective of the size of the tessellated domain. DOE also provided calculations to show that potential caved zones due to seismic events will likely extend to much less than one drift diameter above the drift except for seismic events with an annual exceedance probability of 10^{-6} or smaller.

The NRC staff reviewed DOE's analyses with the expanded boundaries. The additional analyses DOE used provided convincing evidence of continuity in the displacement field crossing the tessellated region, as described in DOE Enclosure 2, Figures 8 and 9 (2010aa). On the basis of the review of DOE's responses to the NRC staff's request for additional information, the NRC staff notes that the use of the original boundary did not significantly affect or limit the estimates of rubble accumulation. In addition, the results of the sensitivity analyses DOE provided demonstrated that with the expanded boundary of the tessellated region, the distressed zone is completely contained within the original (smaller) tessellated domain. Thus, DOE's conclusion is reasonable and rock rubble estimates resulting from seismic events are not underestimated.

Rubble Loading Transmitted to the Drip Shield

DOE used the results of the discontinuum model to estimate the potential drip shield loading due to rubble (SAR Figure 2.3.4-43). DOE also presented alternative bounding analytical approaches for estimating the potential static loading on the drip shields (continuous curves in SAR Figure 2.3.4-46). The analytical model estimates drip shield loading due to rubble using the dead weight of rubble. However, it is well established in the field of rock mechanics that loads transmitted to a drip shield from rubble could differ because of frictional resistance among broken pieces of rock rubble, between rubble and the drift wall, or between rubble and the sides of the drip shield.

To justify using the loads from the numerical model, SAR Section 2.3.4.4.6.3.2 stated that DOE's process-level model accounts for load transmission among rubble particles and to the drip shield. However, DOE in BSC Section P4 (2004al) identified factors that may affect load transmission within rubble and to the drip shield, including size and shape distribution of rubble particles, rubble compaction, and deformability of the drip shield and invert. DOE represented the rock mass in the process-level model as an assemblage of equidimensional polygonal blocks with a characteristic length of approximately 0.2 m [0.65 ft] (SAR Figure 2.3.4-40). SAR Section 2.3.4.4.5.3 and BSC Section 6.4.2.1 (2004al) stated that the model blocks are approximately the same size as potential lithophysal rock blocks.

However, independent NRC staff's analysis in Ofoegbu, et al., p. 4-16 (2007aa) of DOE's fracture data suggests a wide range of rock block sizes and shapes for the lithophysal rock mass, which contrasts with the approximately uniform block size and shape that DOE uses in the rock mass model. Because the potential block size and shape distributions of rubble from the lithophysal rock mass could be different from the approximately uniform block sizes and shapes DOE uses in its drift degradation modeling, the effects of load transmission through rubble could be different from what DOE used in its analysis. Therefore, the NRC staff requested additional information from DOE to demonstrate that appropriate variations in block size and shape have been considered in the UDEC-Voronoi analyses of rockfall during seismic events. In response, the NRC staff received additional information in DOE Enclosure 1 (2010ab) and demonstrated that the bulking factors obtained from UDEC calculations are below the lower end of the ranges for rock rubble because of the assumption that particles are of approximately equal size. This would result in overestimation of the loads acting on the drip shields. DOE conducted additional sensitivity analyses considering different block sizes and demonstrated that, for the range of bulking factors of interest, the average vertical pressure on the drip shield increased only by a small amount (small compared to the standard deviation). Therefore, on the basis of the parametric studies, DOE's conclusion is reasonable and loads were not underestimated when using equidimensional polygons of characteristic length of approximately 0.2 m [0.65 ft].

Summary of NRC Staff's Evaluation of Seismic Drift Degradation

The NRC staff evaluated DOE's assessment of potential degradation of emplacement drifts due to seismic events and estimates of drip shield loading resulting from rubble accumulation. The NRC staff notes that DOE

- Used alternative approaches including empirical, analytical, and numerical models and reasonable methodologies for estimating the timing and extent of drift degradation due to seismic events

- Appropriately considered variability of properties and uncertainties in parameters in the supporting analyses

- Used appropriate rock rubble loads to evaluate performance of engineered barriers

- Justified the abstraction of seismic degradation parameters used for TSPA inputs

2.2.1.3.2.3.4 Drip Shield Structural/Mechanical Performance in the Context of Its Seepage Barrier Function

In DOE's engineered barrier system, the drip shield is a freestanding structure that surrounds the waste package and rests on the crushed rock that forms the invert at the base of the drift. The drip shield is designed to protect the waste package from contact by seepage water and rockfall (SAR Section 2.3.4.5.1.1). The main structural elements of the drip shield are a framework consisting of a bulkhead and support beams (legs) that will be made of Titanium Grade 29. Plates of Titanium Grade 7 are welded onto the framework to form a full composite structure in response to mechanical loading (SAR Section 2.3.4.5.1.1; SAR Figure 2.3.4-56).

Damage to the drip shield can occur from mechanical impacts of falling rocks, by loads from accumulated rock rubble that can be increased by seismic accelerations, and by corrosion processes. Through time, DOE expects that thinning of drip shield components will decrease the capacity of the drip shield to withstand loads and that the likelihood of the drip shield having experienced a potentially damaging load will increase (e.g., SAR Section 2.3.4.1). During the initial few thousand years following repository closure, temperatures within the drifts decrease from around 160 °C [320 °F] to below the boiling point. At these elevated temperatures, generalized corrosion could occur if water contacts the surface of the waste packages. Thus, if the barrier capability of the drip shield fails in the first 12,000 years following repository closure, seepage water could contact the waste package and lead to localized corrosion. DOE relied on the presence of the drip shield as a barrier to preclude significant occurrences of localized corrosion (e.g., SAR Section 2.1.2.2.6).

Consistent with the guidance in the YMRP, the NRC staff focused its review on the risk significant aspects of the drip shield performance. On the basis of DOE's reliance on the drip shield in the demonstration of multiple barrier performance (SAR Section 2.1.2.2.6), the NRC staff focused on evaluating the performance of the drip shield as a barrier to seepage during the early years following drift closure.

For seepage to contact a waste package, openings must occur on the drip shield with sufficient size to permit the advective flow of water through the drip shield plates. Crack openings, such as those produced by stress corrosion , are too small to allow advective flow of water through the drip shield and are excluded from the performance assessment analysis, as described in SNL (2008ab) for FEP 2.1.03.10.0B. Openings large enough for advective water flow could potentially occur through (i) corrosion processes, (ii) impacts of large rock blocks causing puncture of the plates, (iii) physical separation between adjacent drip shield segments due to ground motions from seismic events, (iv) fault displacements, and (v) rupture by deformations that produce effective strains greater than the failure strains in the plates (SNL, 2007ap).

The NRC staff's evaluation of the drip shield corrosion processes is presented in TER Section 2.2.1.3.1. In that review, the NRC staff determined that DOE used reasonable corrosion rates for titanium alloys. Thus, the timing and degree of drip shield component thinning due to corrosion is reasonable for use for mechanical analyses of the drip shield performance in this TER section. DOE excluded large-block impacts from the TSPA as part of the screening analysis for FEP 1.2.03.02.0B (SNL, 2008ab). The NRC staff reviewed the DOE screening arguments in TER Section 2.2.1.2.1 and noted that DOE had appropriately screened out this feature, event, and process from the performance assessment analysis. Thus, the NRC staff's detailed review focused on DOE's representation of processes affecting drip shield separations caused by seismic events or fault displacements, and on the potential for plate rupture.

Separations from Seismic Shaking

Unlikely or low probability seismic events can create ground motions that may cause adjacent drip shields to separate. Consequently, DOE assessed the potential for drip shield separations during seismic events, as described in SNL Section 6.7.3 (2007ay). DOE determined that ground motions large enough to cause potential drip shield separations also cause partial to complete collapse of the repository drifts. DOE determined that rockfall associated with drift collapse occurs during the first seconds of large seismic events. DOE modeled the effect of this rockfall on the ability of the drip shields to separate during seismic events and concluded that rockfall loads from partial drift collapse are sufficient to prevent horizontal separation of the drip shields, as outlined in SNL Section 6.7.3 (2007ay). While these models calculated that minor amounts of vertical separation might occur between the drip shield sections due to settling of the invert or framework damage, the 26-cm [10.24-in]-wide overlaps between the drip shield connectors prevent rockfall or seepage water from contacting the waste package through relatively small vertical separations. In FEP 1.2.03.02.0A (SNL, 2008ab), DOE concluded that seismically induced separations of drip shields can be excluded from the TSPA analysis on the basis of low probability.

NRC Staff's Review of Separations Caused by Seismic Shaking

The NRC staff reviewed the information presented in SNL Section 6.7.3 (2007ay) and analyses in BSC Section 5.3.3.2.2 (2004bq). DOE used rockfall rubble loading conditions that were consistent with the results documented in BSC (2004al) and were appropriate for the seismic events modeled. The NRC staff notes that the DOE model uses the same approach to evaluate the dynamic response of drip shields as was used to evaluate the dynamic response of waste packages. This modeling approach was reasonable as documented in this TER chapter. The NRC staff reviewed the dynamic analyses in SNL Section 6.3.7.1 (2007ay) and notes that potential separations of the drip shield only occurred in an open drift that was subjected to 5.35 m/sec [17.55 ft/sec] peak ground velocity ground motion. NRC staff determined that complete drift collapse is expected for large magnitude seismic events (BSC, 2004al). The NRC staff notes that the DOE approach of modeling an open drift (i.e., no rockfall rubble) would maximize the potential for drip shield separations to occur during seismic events, because the presence of rockfall rubble effectively pins the drip shield segments and restricts the ability for segments to separate. The NRC staff notes that the DOE modeling approach to represent drip shield kinematics during seismic events maximized the potential for separations to occur. DOE appropriately excluded drip shield separations from seismic events in the TSPA analysis.

Separations Caused by Fault Displacement

DOE concluded that fault displacement occurs concurrently with the ground motion during low probability seismic events (SAR Section 2.3.4.5.5.1) and determined that only engineered barrier system components located directly above the moving faults are subject to damage. In the analysis of the effects of fault displacements on engineered barrier system performance, DOE assumed that the drip shield fails completely if fault displacements are sufficient to breach the underlying waste package (SAR Section 2.3.4.5.5.4). In this analysis, DOE assumed that all seepage water entering the drift passes through the failed drip shield, with no diversion of the water.

NRC Staff's Review of Separations Caused by Fault Displacement

The NRC staff's review of the fault displacement model with regard to waste package failure is presented in this TER chapter. DOE's assumption of drip shield failure is reasonable for use in the fault displacement analysis, because the magnitude of a faulting event required to damage a waste package would be sufficient to damage the drip shield. DOE's assumption that a damaged drip shield has no barrier capability is reasonable for use in the performance assessment, because this assumption provides a credible upper bound to the potential significance of fault displacement on drip shield performance.

Plate Rupture by Deformation

To affect performance significantly, the drip shield barrier must fail and allow advective water flow to contact the waste package during the first approximately 12,000 years of postclosure, when environmental conditions are calculated to support localized corrosion of the waste package (e.g., SAR Section 2.1.2). On the basis of the appropriate use of titanium alloy corrosion rates (i.e., TER Section 2.2.1.3.1), negligible thinning of the drip shield components is expected during the first 12,000 years of postclosure. Deformation of the drip shield plates can occur if the underlying framework buckles or collapses due to physical loading. Rupture of the drip shield plate can occur if the magnitude of effective strain on the plate exceeds the strain threshold for the Titanium Grade 7 plates (e.g., SAR Section 2.3.4.5.1.2.2).

NRC Staff's Evaluation of Plate Rupture

The NRC staff's evaluation of the potential for drip shield plate rupture focuses on DOE's modeling approach to evaluate effective stresses in the plates following framework collapse and the basis to determine the location of potential ruptures on the drip shield. The NRC staff focused its review on plate ruptures in the crown area rather than the sides of the drip shields because the ruptures that occur on the sides of the drip shield have negligible potential to allow water contact with the waste package, whereas ruptures on the crown area of the drip shield are likely to allow seepage water to contact the waste package.

Temperature Effects

Mechanical analyses of drip shield performance are dependent on the material properties used in the numerical models. DOE used mechanical properties for the drip shield plates and framework derived from standard handbooks and manufacturer's catalogs (SAR Section 2.3.4.5.1.3.1 and Table 2.3.4-28). DOE considered that a reference temperature of 60 °C [140 °F] for these properties was appropriate, because this temperature is representative of most of the repository closure period. Although DOE recognized that temperatures as high as 300 °C [572 °F] could potentially occur soon after repository closure, DOE considered that the duration of elevated temperatures was too short to warrant consideration for drip shield performance (SAR Section 2.3.4.5.1.3.1).

DOE provided additional information to assess the potential effects of temperatures at or greater than 120 °C [248 °F] on titanium alloy material properties, as discussed in DOE Enclosure 7 (2009bp). During the first 650 years of repository closure, DOE concluded that drip shield temperatures could range from 120–300 °C [248–572 °F]. DOE expects the effect of this temperature increase would not affect titanium alloy properties significantly, because the likelihood for potentially damaging rockfall or seismic events is sufficiently low to preclude significance in the performance assessment. For temperatures below 120 °C [248 °F], DOE compared expected changes in materials properties (e.g., yield strength, tensile strength) to assess the effects of component thinning on the likelihoods of drip shield plate or framework failure. Using small rockfall loads, DOE concluded that changes in the titanium mechanical properties between 60–120 °C [140–248 °F] are a factor of 3 to 4 less than the corresponding percentage changes in component thicknesses that have no significant effect on fragility values.

NRC Staff's Evaluation of Temperature Effects

The NRC staff reviewed the mechanical properties DOE used for titanium alloys at 60 °C [140 °F]. The NRC staff compared these values with values available in standard reference handbooks and noted that DOE used the appropriate mechanical material properties (e.g., yield stress and ultimate tensile strength) for drip shield performance at 60 °C [140 °F].

The NRC staff evaluated the rationale DOE provided to exclude consideration of temperatures greater than 120 °C [248 °F] on titanium material properties. The NRC staff notes that seismic events with $<10^{-5}$ annual probabilities of exceedance are required to produce appreciable amounts of rockfall and that annual probabilities of $<10^{-6}$ are required for reasonable likelihoods of complete drift collapse (SAR Section 2.3.4.4.5.4). The likelihood of appreciable drift collapse occurring in the first 650 years of closure is small. The NRC staff also conducted independent confirmatory analyses that evaluated the effects of increasing temperature from 150 to 260 °C [302 to 500 °F] (Ibarra, et al., 2007aa). These analyses showed an approximately 20 percent decrease in drip shield capacity associated with this temperature increase. This decrease in capacity would not affect the performance significantly, because temperatures associated with this decrease would persist only for hundreds of years and loads associated with potential damage to the drip shield are unlikely to occur.

The NRC staff determined that complete drift collapse from seismic events could occur during the first 12,000 years of repository closure and that the DOE information did not wholly address the uncertainties associated with rockfall load and temperature effects. Consequently, the NRC staff reviewed additional information provided in response to an RAI in DOE Enclosure 7 (2009bp) to evaluate potential temperature effects at 120 °C [248 °F]. Using the methods DOE developed to evaluate 10 percent of rockfall loads, the NRC staff extended this approach to 100 percent of potential rockfall loads. Using the information in SAR Table 2.3.4-43, the NRC staff notes that an approximately 30 percent reduction in drip shield plate thickness minimally increases the likelihood of plate rupture from approximately 1 percent to approximately 5 percent. Although some strength properties show 30 percent variations from 60 to 120 °C [140 to 248 °F], these variations are expected to have only a small to negligible effect on the likelihood of plate rupture for 100 percent collapsed drifts. This is because the titanium plate will have increased ductility and, thus, increased its ability to accommodate deformation without rupture under loads associated with unlikely seismic events. In addition, the NRC staff considers the loads that could potentially increase the likelihood of plate rupture are associated only with earthquakes having $<5 \times 10^{-7}$ annual likelihoods. Using insights from the TSPA, potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. The NRC staff notes that DOE's sensitivity analyses as presented in DOE (2010ac) are reasonable for use to demonstrate that potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. DOE's use of titanium alloy material properties at 60 °C [140 °F] is reasonable for use to evaluate postclosure repository performance, because uncertainties associated with potentially higher temperatures would not significantly affect the results of the performance assessment.

Modeling Approach

To evaluate drip shield plate capacity, DOE conducted numerical modeling of the drip shield under quasi-static and dynamic loading conditions. For the quasi-static analyses, DOE calculated rock rubble loads on the drip shield, multiplied these loads by the vertical component of peak ground acceleration, and modeled the drip shield response to these loads. DOE calculated the quasi-static loading conditions on the drip shield plate using FLAC3D, a three-dimensional finite-difference computer code. DOE calculated stresses and strains on one-half of the plate on the drip shield crown, which represents one segment between two framework bulkheads.

DOE used rock rubble loads calculated from UDEC analyses of rockfall during seismic events (SAR Section 2.3.4.5.3.2). DOE evaluated two static loading configurations on the drip shield. One configuration used an average of six UDEC realizations for each modeled segment of the drip shield, which DOE used to consider spatial variability in the nonuniform load. The second configuration used a single UDEC realization, which DOE considered as representative of the highest loads on the drip shield crown (SAR Section 2.3.4.5.3.2.1).

For each loading configuration, the vertical load was applied over the entire top surface of the plate and increased incrementally until a failure mechanism developed, as described in SNL Section 6.4.3.1.2 (2007ap). For each load increment, the model compared the residual tensile stresses or accumulated plastic strain against a failure criterion of 80 percent of the yield strength for Titanium Grade 7, as outlined in SNL Section 6.4.3.1.3 (2007ap). DOE concluded that plate failure occurred at the smallest applied load that exceeded either the stress or strain criterion.

By uniformly increasing the static load in the UDEC model, DOE calculated that an intact drip shield plate has a capacity (i.e., limit load) of approximately 2,500 kPa [52,218 psf], which is approximately twice the calculated capacity of the drip shield framework (SAR Section 2.3.4.5.3.3.1). To determine the likelihood of drip shield plate failure, DOE integrated the annual likelihood of exceeding levels of ground acceleration with the likelihood of rupture for plates experiencing the loads corresponding to the level of ground acceleration. For intact drip shield plates and 100 percent rockfall load, DOE calculated that seismic events with annual probabilities of exceedance $<5 \times 10^{-7}$ can lead to plate rupture on 1–7 percent of the drip shields (SAR Section 2.3.4.5.3.4).

As an alternative to the quasi-static analyses, DOE also conducted dynamic analyses for drip shield plate capacity using the UDEC computer code (SAR Section 2.3.4.5.3.3.3). These analyses used a two-dimensional cross section of the drip shield surrounded by rock rubble. The dynamic analyses used vertical ground accelerations from time histories that DOE views as representative of larger magnitude seismic events in the Yucca Mountain region. DOE applies these vertical accelerations to the basal boundary of the UDEC model, which allows the emplacement drift, rubble, and drip shield to interact dynamically for the modeled period of strong ground motion. DOE compared the results of the dynamic analyses with the quasi-static analyses and concluded that the quasi-static model underestimates the stability of the drip shield plates. DOE therefore concluded that the quasi-static approach is an appropriate basis to calculate the likelihoods of plate rupture, as the dynamic analysis would result in lower likelihoods of rupture (SAR Section 2.3.4.5.3.3.3).

NRC Staff's Evaluation of Modeling Approach

The NRC staff reviewed the use of the FLAC3D computer code in the analyses of the drip shield plate capacity. The NRC staff reviewed the information in SNL Section 7.3.3.1 (2007ap) and determined that DOE appropriately compared the FLAC3D model results with an alternative approach used in structural mechanical analyses (i.e., LS-DYNA). In response to the NRC staff's request for additional information (RAI), DOE provided the additional information in DOE Enclosure 8 (2009bp) to address the representation of nonlinear responses of materials. On the basis of review of this information, DOE provided appropriate support for use of the FLAC3D computer code to calculate drip shield plate performance.

The NRC staff notes that the rock rubble static loads used in the models were consistent with the degraded drift configurations used elsewhere in SAR Section 2.3.4.5 for seismic events, and that the UDEC analysis representation of this rubble resulted in bulking factors that were appropriate for the Topopah Springs lithophysal tuff.

The NRC staff conducted independent confirmatory calculations using an alternative modeling approach to evaluate drip shield deformation from rock loading (Ibarra, et al., 2007aa). The NRC staff compared the deformation patterns determined from the independent model to the deformation patterns DOE determined using the dynamic modeling approach. This comparison showed that because the drip shield framework has lower capacity than the plates, deformation is most likely to occur on the legs of the drip shield and not on the crown. The NRC staff notes that the quasi-static analyses provide a reasonable basis to determine the likelihood of drip shield plate rupture from seismically accelerated rock rubble loads.

Drip Shield Framework Deformation

DOE calculated the likelihood of drip shield framework failure using the same approach as implemented for the drip shield plate analyses (SAR Section 2.3.4.5.3.3.2). These analyses determined that the drip shield framework has approximately half the bearing capacity as compared to the drip shield plates and that buckling of the drip shield legs results from exceeding the bearing capacity. DOE also determined that if the drip shield becomes tilted after the framework buckles, the drip shield connector plate and connector guide provide a physical barrier that will divert seepage from the crown to the sides of the drip shield, as outlined in SNL Section 6.7.3.2 (2007ay).

DOE postulated that if one segment of a drip shield collapsed more extensively than adjacent segments, localized stresses may lead to rupture of the drip shield plates along the crown. DOE considered the likelihood of isolated segment collapse to be low, because rubble loads are expected to be relatively uniform and the rigidity of the drip shield is expected to effectively transfer loads to the adjacent segments (DOE, 2010ac). Thus, DOE expects complete collapse of the drip shield when loads exceed the design capacity. Nevertheless, DOE analyzed stress-strain relationships for a partially collapsed drip shield and determined that plate rupture would occur if vertical displacements between adjacent segments exceeded approximately 19 cm [7.5 in] (DOE, 2010ac). DOE concluded that such displacements between adjacent segments are unlikely to occur, because the structure of the drip shield will effectively transfer stress from a deforming segment onto the adjacent segments. This stress transfer leads to a progressive collapse of adjacent drip shield segments, rather than isolated collapse and potential tearing of a single segment (DOE, 2010ac).

NRC Staff's Evaluation of Drip Shield Framework Deformation

The NRC staff conducted independent confirmatory calculations using an alternative modeling approach to evaluate drip shield deformation from rock loading (Ibarra, et al., 2007aa). These calculations confirmed that buckling of the drip shield framework is expected in the legs. The NRC staff evaluated the drip shield design DOE provided (e.g., SAR Figure 1.3.4-15). On the basis of the low volumes of seepage water potentially contacting the drip shield, the NRC staff noted that the drip shield connector plate and connector guide adequately divert seepage water from the crown area if the drip shield is tilted due to framework buckling. Thus, uncertainties in the drip shield framework capacity would not affect the potential for seepage water to contact the waste package through tilting of the drip shield due to buckling of the drip shield legs.

The NRC staff notes that potential underestimation of the drip shield framework capacity would result in early transitions to a damage state (i.e., SAR Section 2.3.4.5.4.1, Idealized Damage State 2) where the waste package is pinned by the collapsed drip shield. As a result, the waste package would be restrained from movement during large magnitude seismic events and have a reduced potential for stress corrosion cracking. For unrestrained motion during seismic events, as would occur when the drip shield is intact, up to 4 percent of the waste package surface area can be damaged sufficiently for stress corrosion cracks (SCC) to develop (SAR Section 2.3.4.5.2.1.4.2). In contrast, a collapsed drip shield localizes the potential damaged area on the waste package and results in an approximately order-of-magnitude decrease in the potential for stress corrosion cracking (SAR Section 2.3.4.5.4.3.2.1). The NRC staff noted that potential uncertainties that result in DOE overestimating drip shield framework capacity are not significant to performance, because increasing the ability of the drip shield framework to withstand seismic loads increases the potential for larger amounts of waste package damage, and potential radionuclide releases, through SCC.

The NRC staff reviewed the DOE modeling approach for evaluating plate response during partial drip shield framework collapse and notes that DOE used appropriate physical parameters and reasonable geometries to evaluate stress-strain relationships (DOE, 2010ac). The NRC staff notes that (i) the DOE modeling approach (using the FLAC3D and LS-DYNA computer programs) was consistent with standard practice for determining stress-strain relationships and (ii) DOE's failure criteria were appropriately implemented.

The NRC staff notes the following DOE assumption is reasonable: relatively uniform rock-rubble loads should occur on a drip shield during seismic events, on the basis of the relatively uniform characteristics of the expected rock rubble and because most drifts are wholly collapsed during potentially damaging, very low probability seismic events. Nevertheless, the NRC staff recognizes that DOE represented heterogeneity in rubble load for some calculations for evaluating drip shield fragility [e.g., SNL Section 6.4.3.2.2.2 (2007ap)]. Thus, some potential exists for localization of rubble loads on the drip shield during very low probability, large magnitude seismic events. As a consequence, the potential exists for the rubble load to be greater on some segments of a drip shield than on adjacent segments. However, in the context of how these results are used in DOE's performance assessment, a more detailed consideration of the potential nonuniform load distribution is not likely to significantly affect the overall structural performance of the drip shield frame. Therefore, the NRC staff notes that such variations have minor effects on the overall seepage barrier performance of drip shields.

On the basis of the review of the drip shield design and the expected response of the drip shield as a composite structure, significant stress redistribution would occur to adjacent drip shield segments, if loading was localized on an individual segment. The NRC staff notes that differential collapse of the drip shield would most likely involve at least several adjacent segments, which would be sufficient to prevent localized strains from exceeding the failure strain of the drip shield plates. The NRC staff recognizes that there is uncertainty in the amount of vertical displacement that the drip shield plates can accommodate {i.e., 18.2 cm [7.17 in] between two bulkheads} before the failure strain criterion derived from an assumed differential displacement of 19.8 cm [7.8 in] might be exceeded, as described in DOE Enclosure 9 (2010ac). At the same time, the NRC staff notes that the analysis DOE provided likely overestimates failure potential because (i) the boundary conditions on the bulkheads are assumed to be fixed (which overestimates stresses) and (ii) the three longitudinal stiffeners are neglected (which underestimates the overall stiffness of the composite structure).

To evaluate the potential significance of the uncertainty in the barrier capability of the drip shield plates during seismic events, the NRC staff used insights from the TSPA analysis for intrusive igneous events (SAR Section 2.4.2.2.1.2.3). In that analysis, DOE considered that an igneous intrusive event removed all barrier capabilities from the drip shield and the waste package, and made all waste available for dissolution and transport. Using an approximately 10^{-8} average annual probability of occurrence, DOE calculated a probability-weighted igneous intrusive dose equivalent of less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period (SAR Section 2.4.2.2.1.2.3.1). In DOE (2009aa) and SNL Appendix P (2008ag), DOE also showed that increasing the average annual probability of occurrence to 10^{-7} increases the expected annual dose equivalent to less than 0.006 mSv/yr [0.6 mrem/yr].

In comparison to an igneous intrusive event, seismic events have limited potential to create openings in drip shield plates that are large enough to permit advective water inflow. In addition, only a limited range of potential seepage waters has compositions that support potential localized corrosion processes (TER Section 2.2.1.3.3), in-drift conditions can support potential localized corrosion processes for only a limited period of time, and potential openings that could result from localized corrosion are small (TER Section 2.2.1.3.1). Thus, potential releases from uncertainties in drip shield performance under seismic conditions would be appreciably smaller than what could be expected for an igneous intrusive event. Given that an igneous intrusive event (which essentially removes all engineered barrier system capabilities) contributes less than a dose equivalent of 0.006 mSv/yr [0.6 mrem/yr] to the total effective dose calculation, uncertainties related to the drip shield barrier performance during seismic events are expected to cause insignificant variations in the total annual dose. Thus, DOE has appropriately accounted for the performance of the drip shield barrier function in the performance assessment, because uncertainties in the DOE evaluation would not affect the results of the performance assessment significantly.

Summary of NRC Staff's Evaluation of Drip Shield Performance

DOE relies on the drip shields as effective barriers to advective water flow or rock rubble impacts on the waste package. The NRC staff reviewed the information DOE presented relevant to the barrier capability of the drip shield and notes the following:

- DOE appropriately identified potential processes that may lead to openings in the drip shield that affect barrier capabilities.

- DOE used reasonable models and information to demonstrate that potential openings from horizontal or vertical displacements during seismic events would not affect performance significantly.

- DOE reasonably assumed that fault displacements sufficient to damage a waste package remove all barrier capabilities from the associated drip shield.

- DOE appropriately evaluated the potential for ruptures in the drip shield plates during the first 12,000 years of closure by taking a conservative approach.

- DOE appropriately determined that a small likelihood exists for such ruptures if earthquakes with annual probabilities of exceedance of $<5 \times 10^{-7}$ occur. DOE appropriately implemented this likelihood of plate failure in the TSPA.

- DOE reasonably demonstrated that uncertainties in this information would not affect the results of the performance assessment significantly.

The NRC staff notes that DOE reasonably evaluated the barrier capabilities of the drip shield during mechanical deformation events and has appropriately incorporated the risk-significant aspects of this evaluation into the performance assessment calculations.

2.2.1.3.2.3.5 Waste Package Mechanical/Structural Performance

The DOE performance assessment includes information related to the mechanical disruption of engineered barrier abstractions and calculates waste package performance. DOE classified the waste package as important to waste isolation (SAR Table 2.1-1). DOE provided information on structural response of the waste package to mechanical disruption in SAR Section 2.3.4.5. The objective of this TER section is to evaluate whether reasonable technical bases have been provided for waste package abstractions used in DOE's performance assessment.

DOE assessed potential waste package mechanical damage by performing detailed structural analyses. The results of these structural analyses were used as inputs to the seismic consequence abstractions (SCA). The SCA simulates mechanical interactions among the waste packages, the drip shield, the emplacement pallet, and/or accumulated rubble as a function of peak ground velocity. DOE calculated waste package damage as (i) stress corrosion cracks (SCCs) that may allow diffusive radionuclide releases and (ii) rupture and puncture areas that may allow advective radionuclide releases (reviewed in TER Section 2.2.1.3.4.3.5).

The results of the seismic consequence abstractions are used as inputs to other process-level models and direct inputs to the TSPA. The waste package corrosion abstraction uses waste package breaches at the process level to initiate double-sided corrosion (reviewed in TER Section 2.2.1.3.1). Note that in this context, a breach is defined as any failure mechanism that penetrates the waste package (i.e., cracks, ruptures, and punctures). Waste package breaches also impact the chemistry inside the waste package (reviewed in TER Section 2.2.1.3.4). Stress corrosion crack area is used in the engineered barrier system transport abstraction to model a pathway for diffusive radionuclide release (reviewed in TER Section 2.2.1.3.4). Waste package rupture or puncture area is used in the flux-splitting model to calculate water flux through the waste package (reviewed in TER Section 2.2.1.3.3).

Information presented in SAR Table 2.1-3 suggests that seismic ground motion damage to the engineered barrier system components is an important mechanism that affects the engineered barrier system capability to perform its intended functions. DOE stated in DOE Enclosure 1 (2009bl) that seismic-induced waste package damage is more significant in early times and that nominal failure processes are more significant at later times. According to DOE, seismic-induced stress corrosion cracking is the most probable waste package damage mechanism. The majority of commercial spent nuclear fuel (CSNF) and codisposal waste package failures due to seismic-induced stress corrosion cracking occur prior to drip shield plate/crown failure, as described in DOE Enclosure 1, Figures 5 and 6 (2009bl).

As described next, DOE considered three idealized states of the engineered barrier system (SAR Section 2.3.4.5):

1. Structurally stable drip shield state (intact drip shield)—when the waste packages are free to move and may be damaged due to impacts with other engineered barriers during seismic events

2. Drip shield framework failure state (collapsed drip shield)—when the drip shield–waste package interactions during seismic events may damage the waste package outer barrier

3. Drip shield plates failure state—when the waste package is surrounded by and in direct contact with rubble and may be damaged due to waste package–rubble interactions during seismic events

As DOE detailed in DOE Enclosure 1, Figure 1 (2009bl), nominal stress corrosion cracking in a commercial spent nuclear fuel waste package would initiate between 200,000 and 300,000 years, when the timeframe is dependent on the drip shield performance (reviewed in TER Section 2.2.1.3.2.6). This would happen after the beginning of Idealized State 2. The commercial spent nuclear fuel waste packages cannot move as freely in Idealized State 2 as in Idealized State 1, thereby reducing the potential for seismic induced stress corrosion cracking.

For the three idealized states, DOE considered two waste package failure modes.

1. The first failure mode is referred to as “the residual stress failure mode” in this TER section. The waste package damage is expressed in terms of the waste package outer corrosion barrier surface area that may be susceptible to stress corrosion cracking. It is defined as an area with the residual stresses exceeding one of three residual stress threshold values: 90, 100, and 105 percent of the Alloy 22 yield stress (reviewed in TER Section 2.2.1.3.1.3.2.3).

2. The second failure mode is referred to as “the tensile tearing failure mode” in this TER section. DOE used Alloy 22 ultimate tensile strain as a failure criterion to evaluate the waste package outer barrier tensile tearing (rupture and/or puncture) occurrence.

For these two failure modes, DOE developed the abstractions using a three-part approach: (i) the rupture/puncture probability was defined as a function of peak ground velocity and the effective tensile stress limits; (ii) the probability of a nonzero damaged area was defined as a function of peak ground velocity and the residual stress threshold damage; and (iii) for nonzero damaged area cases, a conditional probability distribution for the magnitude of the conditional damaged area was defined as a function of peak ground velocity and the residual stress threshold.

DOE's analyses results indicate greater mechanical damage potential to the waste package during Idealized State 1. However, the NRC staff reviewed the fundamental aspects of damages in all three idealized states and their abstractions. The review presented in this section is organized around these major topics considering the context of DOE's performance assessment.

Idealized State 1: Waste Package Structural Response with Structurally Stable Drip Shield

Modeling Assumptions and Approach

In SAR Section 2.3.4.5.2.1, DOE provided information on waste package structural response for the Idealized State 1 where the drip shield is structurally stable. DOE considered that dynamic impacts of the waste package on the rest of the engineered barrier system components may lead to waste package damage and rupture of the outer corrosion barrier. DOE evaluated the movement of and damage to waste packages resulting from seismic loads. The following three cases of impacts were considered using numerical models: (i) impacts between waste packages, (ii) impacts between the waste package and the emplacement pallet, and (iii) impacts between the waste package and the drip shield (SAR Section 2.3.4.5.2.1). DOE analyzed the transportation and disposal (TAD) and the codisposal (CDSP) waste packages for three waste package conditions where the drip shield is expected to remain functional and structurally stable (SAR Section 2.3.4.5.2.1). The three conditions are (i) 23-mm [0.91-in]-thick outer corrosion barrier with intact internals; (ii) 23-mm [0.91-in]-thick outer corrosion barrier with degraded internals; and (iii) 17-mm [0.67 in]-thick outer corrosion barrier with degraded internals. (DOE modeled a waste package with degraded internals as the waste package outer corrosion barrier only.)

The NRC staff reviewed these three waste package conditions DOE analyzed using guidance in YMRP Section 2.2.1.3.2.2. As mentioned earlier, DOE considered 23- and 17-mm [0.91- and 0.67-in] waste package outer corrosion barriers. These represent corrosion thinning of 2.4 and 8.4 mm [0.09 and 0.33 in], respectively, from the initial 25.4-mm [1-in] outer corrosion barrier thickness and correspond to a timeframe of approximately 340,000 and 1.2 million years after emplacement (reviewed in TER Section 2.2.1.3.1.3.2.1). The time duration DOE considered covers the period of interest (i.e., 1 million years) for the mechanical disruption of engineered barriers. Therefore, DOE appropriately addressed uncertainties in the waste package conditions and environmental effects on the waste package components. Consideration of uncertainty was accomplished through appropriate reductions of the waste package outer corrosion barrier thickness and degradation of the waste package internals.

Using guidance in YMRP Section 2.2.1.3.2.2, the NRC staff reviewed the material proprieties of the engineered barrier system components DOE incorporated into the numerical models. The NRC staff compared the values of these mechanical properties with the information available in the open literature (American Society of Mechanical Engineers, 2001aa) and determines that DOE used appropriate values of the mechanical properties for engineered barrier system components.

The NRC staff notes that DOE did not include the waste package damage potential for impact between the waste package and drip shield in the seismic damage abstractions. DOE's decision was based on the observations of the waste package damage from the analyses of impacts between waste packages (SAR Section 2.3.4.5.2, p. 2.3.4-131). DOE concluded that the waste package areas damaged from a side impact on a flat elastic surface were zero or very small. These damaged areas were significantly less than the damaged areas from end impacts, as described in SNL Table 6-13 (2007ay). DOE stated that the waste package side impacts on a flat elastic surface are representative of the waste package impacts on the drip shield side wall.

NRC Staff's Evaluation of Modeling Assumptions and Approach

The NRC staff reviewed this assumption using YMRP Section 2.2.1.3.2 and noted that DOE's assumption is well supported and the results from impacts between waste packages bound the results for side, or lateral, impacts between the waste package and the drip shield. Thus, DOE's basis for stating that the waste package damage due to side impacts with a drip shield caused by seismic events was bounded by considering the end impacts between waste packages.

DOE also stated that vertical impacts between the waste package and the drip shield would have a small contribution to the total waste package damage (SNL, 2007ay). DOE concluded that the impact loads on the waste package would be distributed over a large contact area of the drip shield bulkheads and stiffeners. DOE further concluded that vertical impacts between the waste package and the drip shield surrounded by rubble would be similar to impacts between waste packages, which also result in small damaged areas. Therefore, DOE concluded that the impact damage between waste packages is representative of the waste package damage from vertical impacts between the waste package and drip shield.

The NRC staff reviewed these assumptions and noted that the vertical impact between the waste package and the drip shield would be similar to impacts between the waste packages and the pallets. The NRC staff also reviewed information presented on the frequency of the vertical impacts between the waste package and the drip shield in SNL Section 6.4.5 (2007ay). On the basis of this information, in 17 realizations of kinematic analyses at peak ground velocity levels of 1.05 and 2.44 m/sec [3.44 and 8 ft/sec], the number of impacts between the waste package and the drip shield bulkheads was less than 10 and at peak ground velocity levels of 4.07 m/sec [13.35 ft/sec] the number of impacts increased to 48. Note that the annual probability of exceedance of a seismic event associated to the peak ground velocity of 4.07 m/sec [13.35 ft/sec] is on the order of 10^{-8}. Because the frequency of occurrence for the vertical impacts between the waste package and the drip shield is low, the waste package damage due to these impacts would be small when compared to the waste package damage from waste package to waste package and waste package to pallet impacts. Therefore, the assumption DOE used would not significantly affect TSPA results.

To estimate waste package damage and rupture potential, DOE developed a two-part calculation process using numerical models developed in the computer code LS-DYNA (Livermore Software Technology Corporation, 2003aa).

First, large-scale kinematic analyses were performed to determine the impact parameters for multiple waste packages in an emplacement drift. The parameters included locations and time of impacts, relative velocity and impact angles, and forces between the impacting bodies. DOE used 17 ground motion time histories at peak ground velocity levels of 0.4, 1.05, 2.44, and 4.07 m/sec [1.31, 3.44, 8, and 13.35 ft/sec] in these analyses. The NRC staff reviewed these values using YMRP Section 2.2.1.3.2 and notes that the ground motions used are consistent with the values presented on the bounded hazard curve in SAR Figure 2.3.4-18. The large-scale kinematic calculations presented in the SAR consider a “string” of multiple waste packages. A combination of TAD and codisposal waste packages in a section of an emplacement drift was considered. For these analyses, DOE considered a partially or fully collapsed emplacement drift. The drip shield was considered to be in a structurally stable condition. Thus, the structurally stable drip shield provided the only restriction to the movement of the waste packages and the pallet. DOE recorded impacts for the central waste packages (three and two central waste packages for the TAD and codisposal configurations, respectively) in the total string of waste packages (SAR Section 2.3.4.5.2.1.3.1).

Second, DOE carried out detailed finite elements analyses for estimating damage and rupture potential. Impacts between individual waste packages and between waste package and pallet were analyzed. DOE evaluated waste package damage over the range of impact parameters, including those determined from the large-scale kinematic analyses. Using the results of the detailed finite element analyses, DOE estimated the waste package damage and rupture potential for the multiple impacts modeled using the large-scale kinematic analyses.

Using guidance in YMRP Section 2.2.1.3.2, the NRC staff reviewed the modeling approach DOE employed to evaluate the waste package response to vibratory ground motions while the drip shield is structurally stable. DOE followed established industry practice for performing finite element analyses (Bathe, 1996aa) of mechanical/structural components of the waste package. The NRC staff considers reasonable and appropriate the geometric representation of the waste package and its components and associated simplifications made to the waste package geometry. The finite element models were appropriately used for characterizing waste package damage as input to the TSPA calculations. Further, the results for the central waste packages from a string of waste packages would be representative and would not be affected by the model boundaries along the emplacement drift direction.

DOE stated that the waste package pallet eventually fails as the stainless steel connector tubes lose their structural integrity (SAR Section 2.3.4.1). For the damage analyses, however, DOE made an assumption that the waste package pallet is intact. This assumption, according to DOE, would lead to greater damage to the waste package outer corrosion barrier during vibratory ground motion. As DOE explained, the reason for this conclusion is that higher magnitude stresses are generated when the waste package impacts a “relatively stiff pallet as opposed to the crushed tuff invert” (SAR Section 2.3.4.1, p. 2.3.4-10). However, DOE initially did not consider that loss of connector integrity could result in a different range of conditions for pallet pedestal orientations, impact locations, and impact frequencies. The NRC staff questioned whether the variability in these parameters may have exceeded the range DOE considered. The NRC staff’s review indicated that larger uncertainty in the pedestal orientation can potentially affect the calculated results. For example, impact locations, time of impact, relative velocity of the impacting bodies, angle of impacts, and forces between the impacting bodies could be affected. To clarify this question, DOE was requested to supplement the information presented in SAR Sections 2.3.4.5.2 and 2.3.4.5.4 to address whether such uncertainties would affect significantly the characteristics of waste package damage calculated in kinematic analyses.

In response to the NRC staff’s request for additional information, DOE provided additional evaluation in DOE Enclosure 1 (2009bq) to demonstrate that the intact waste package pallet assumption did not underestimate the potential for waste package damage in the kinematic analyses. DOE stated that at lower peak ground velocity levels, the waste package and the pallet pedestals would have limited relative motion. Therefore, if the stainless steel connector tubes were to lose structural integrity due to corrosion, the waste package damage would remain bounded by the results of the analyses with the intact waste package pallet. DOE stated that, at higher peak ground velocity levels and degraded connector tubes, the impact between the waste package and pallet would be characterized by one of three cases: the waste package impacts both pallet pedestals (Case 1), the waste package impacts one pallet pedestal (Case 2), and the waste package impacts only the invert (Case 3).

For Case 1, DOE stated that the angles and locations of impacts would be similar to those used in the kinematic analyses with an intact pallet. For Case 2, DOE stated that the locations of the impacts would be toward the end of the waste packages, as the waste package would tend to slide off the remaining pedestal and onto the invert. In SNL Tables 6-49 and 6-50 (2007ap) DOE stated that, due to higher waste package stiffness at the waste package lid, the waste package would experience less damage for impacts near the waste package lids than in the middle of the waste package. DOE concluded that for Case 2, the waste package damage would be bounded by the results of the analyses for waste package impacts with an intact pallet. For Case 3, DOE stated that the waste package damage would be bounded by the results of the intact pallet. This result is due to the waste package experiencing less damage from impact forces distributed over a larger waste package area. Therefore, DOE concluded that the results of the analyses with an intact pallet would bound the waste package damage for the case of structural integrity loss from corroded stainless steel connector tubes.

The NRC staff reviewed DOE's responses to the staff's RAI and noted that

- The waste package damage from impacts with an intact waste package pallet would bound the waste package damage from impacts with two separated pallet pedestals because the contact area of impact would not change

- The waste package damage from angular impacts with an intact waste package pallet would be representative for the waste package damage with a single pallet pedestal because such an impact causes more damage than other impacts

- The waste package damage from impacts with an intact waste package pallet would bound the waste package damage with the invert because the area of impact would increase and, as a result, reduce the stresses in the waste package outer corrosion barrier

DOE considered waste package damage from angular impacts with an intact waste package pallet and concluded that the waste-package-to-pallet impacts are likely to cause more waste package damage than other types of impacts. The NRC staff's review considered that DOE did not address the potential change in the frequency of this type of impact due to pallet degradation. However, because of pallet degradation, the number of waste package angular impacts would not significantly increase. Due to the close proximity of the waste packages to each other, and other engineered barrier system components within drip shield boundaries, the likelihood of waste package angular impacts on a single pallet pedestal is only feasible at very high peak ground velocity levels. On the basis of DOE's seismic hazard curves, the NRC staff notes that the annual probability of exceedance of a seismic event associated with such high peak ground velocity levels is very low. Therefore, the DOE modeling approach is appropriate for use in the TSPA calculations.

Residual Stress Failure Mode

To analyze the residual stress failure mode, DOE calculated the total damaged area of the waste package. Total damaged area is defined as the sum of the areas of all outer corrosion barrier elements in which the stress exceeds a threshold stress level at the end of a simulation. The three residual stress threshold values used are 90, 100, and 105 percent of the yield strength. (The NRC staff's evaluation of the residual stress threshold values DOE used to estimate the waste package damaged area is presented in TER Section 2.2.1.3.1.3.2.3.)

DOE used results from the analyses of the impacts between waste packages and between the waste package and the pallet. Inputs for the TSPA calculations were prepared in the form of lookup tables that provided damaged area as a function of the impact parameters. According to the information provided in these lookup tables (SNL, 2007ap), the amount of damage for single impacts is largest for impacts between a waste package and a pallet. The damage increases with a decrease in the outer corrosion barrier thickness. The reported damage area for single impacts ranged from 0.002 to 14.333 percent of the total surface area for the TAD waste package and from 0.002 to 20.106 percent for the codisposal waste package.

For the analyses with multiple waste packages, the amount of reported damage is largest for impacts between waste a package and a pallet. The damage increases with an increase in peak ground velocity levels and a decrease in the outer corrosion barrier thickness. The reported damage area ranged from 0.006 to 43.467 percent of the total surface area for the TAD waste package. The range for the codisposal waste package used by DOE was from 0.006 to 19.585 percent of its surface area (SNL, 2007ap).

NRC Staff's Evaluation of Residual Stress Failure Mode

The NRC staff reviewed the residual stress failure mode results for these analyses and considered DOE's response to an RAI (DOE, 2009br) that addressed whether the intact waste package pallet assumption would not underestimate the potential for waste package damage in the kinematic analyses. DOE followed established industry practice in performing these finite element analyses, incorporated reasonable simplification and defensible assumptions, and used appropriate loading conditions to characterize the waste package damage. Therefore, the waste package damage results for the residual stress failure mode are reasonable for use as input to TSPA analyses.

Tensile Tearing Failure Mode

To analyze the tensile tearing failure mode, DOE assessed the rupture condition for a single impact. The maximum effective strain in the waste package outer corrosion barrier for the full time-history analyses was compared with the rupture tensile strain failure criterion (SAR Section 2.3.4.5.2.1.3.2). DOE demonstrated through detailed finite element calculations that the strain for a single impact in the outer corrosion barrier was always below the ultimate tensile strain for Alloy 22 (SAR Section 2.3.4.5.2.1.3.2). For multiple impacts modeled in the large-scale kinematic analyses, DOE stated that if an impact causes "severe" deformation, the additional large impacts to the deformed area have the potential to cause rupture. For both the TAD and the codisposal waste packages with intact internals, DOE stated that the overall deformation of the outer corrosion barrier resulting from multiple impacts was insignificant even at the largest impact velocities. Therefore, DOE concluded that no rupture would occur (SAR Section 2.3.4.5.2.1.3.2).

For the analyses with degraded internals, DOE considered that the deformation from low velocity impacts {peak ground velocity levels less than 1.05 m/sec [3.44 ft/sec]} was not severe enough to lead to rupture after multiple impacts. In addition, the deformation becomes very large as the impact velocity increases. For peak ground velocity levels of 1.05 m/sec [3.44 ft/sec] and higher, a second impact of equal or greater magnitude would potentially cause a rupture of the outer corrosion barrier. Therefore, for the peak ground velocity of 1.05 m/sec [3.44 ft/sec] and higher, which have a mean annual probability of exceedance of 10^{-5}, the waste package rupture probability exceeds zero. In some realizations of large-scale models for both the TAD and the codisposal waste package configurations with degraded internals and peak ground velocity levels of 2.44 m/sec [8 ft/sec] and higher, DOE calculated the probability of rupture equal to one.

NRC Staff's Evaluation of Tensile Tearing Failure Mode

The failure criterion (i.e., ultimate tensile strain) DOE used to evaluate the waste package rupture occurrence from a single impact is consistent with industry practice and is widely used in the field of mechanical/structural engineering (American Society of Mechanical Engineers, 2001aa). DOE relied on engineering judgment to determine whether multiple impacts to the waste package result in tensile rupture (SAR Section 2.3.4.5.1.4.2). If the degree of deformation from a single impact was judged significant by DOE, a second impact of equal or greater magnitude was judged sufficient to cause tensile rupture. However, DOE initially did not describe the magnitude of stress or strain on the outer corrosion barrier, the impact velocities that caused this damage, or the threshold beyond which such damage occurs. The NRC staff determined that the SAR did not explain how variations in these or other indicators of damage were considered in the expert judgment process and therefore requested additional information.

In response to the NRC staff's request for additional information, DOE provided information in DOE Enclosure 2 (2009bq) to demonstrate the acceptability of the methodology that involves engineering judgment used in the qualitative evaluation of waste package rupture probability for multiple impacts. DOE performed a quantitative evaluation of the waste package rupture probability. The analysis is based on maximum effective strain limit and assessment of tensile strain in the waste package outer corrosion barrier. Because the quantitative approach did not predict waste package rupture, DOE developed a qualitative approach. This approach was based on an assessment of the outer corrosion barrier deformation. The deformation results were used to estimate the waste package rupture probability for multiple impacts. In SNL Figures 6-31 through 6-36 (2007ap), DOE examined deformation shapes of the outer corrosion barrier to determine a deformed state that could cause rupture if a second large impact occurred. For the analyses at an impact velocity of 5 m/sec [16.4 ft/sec], DOE stated that the outer corrosion barrier developed deformations sufficient to cause rupture at a subsequent seismic event. DOE defined this state as a lower bound such that another impact of 5 m/sec [16.5 ft/sec] or higher would cause rupture of the waste package outer corrosion barrier. DOE used impact force values associated with impacts at 5 m/sec [16.4 ft/sec] as a threshold force to define zero probability of waste package rupture due to multiple impacts. DOE defined the force associated with impacts at 7 m/sec [23 ft/sec] as an "upper force peg point" and used this to interpolate and extrapolate probability of waste package rupture between zero and one. DOE concluded that this qualitative method would not underestimate waste package rupture probability, because the force threshold used as a lower bound was derived on the basis of less severe and more frequent waste package deformations at impact velocities of 5 m/sec [16.4 ft/sec] and higher.

This threshold value is reasonable for use because another impact of 5 m/sec [16.4 ft/sec] or higher would rupture the waste package outer corrosion barrier. The NRC staff notes the following. Waste-package-to-pallet impacts would lead to greater damage to waste packages than the waste-package-to-waste-package or the waste-package-to-drip-shield impacts. Further, for the waste-package-to-pallet impacts, the most damaging scenario is angular impacts at 6° angles into the middle of the TAD waste package with degraded internals (SNL, 2007ap). The NRC staff could not determine the information on peak ground velocity levels that would trigger impact velocities of 5 m/s [16.4 ft/sec] or higher for the waste-package-to-pallet impacts. Therefore, the NRC staff was not able to estimate the probability of occurrence of seismic events triggering these impact velocities. However, DOE provided information on impact velocities of the drip-shield-to-waste-package impacts in SNL Tables 6-148 through 6-150 (2007ap). On the basis of this information, impact velocities of 4 m/sec [13.12 ft/sec] or higher are likely to occur only for seismic events at peak ground velocity levels of 4.07 m/sec [13.35 ft/sec] or higher. The NRC staff estimated that for a given peak ground velocity level, the waste-package-to-pallet impacts would exhibit impact velocities similar to those observed for the drip-shield-to-waste-package impacts. Therefore, impact velocities of 5 m/sec [16.4 ft/sec] are likely to occur only for seismic events with an annual probability of exceedance of 10^{-8} or lower. Thus, subsequent seismic events capable of triggering these large impact velocities are unlikely and therefore beyond consideration for TSPA analyses. Moreover, the NRC staff also notes that the waste package should have enough remaining capacity, after the first seismic event with impact velocities of 5 m/sec [16.4 ft/sec], to withstand subsequent seismic events at these impact velocities. This is based on DOE-provided numerical results indicating the waste package would not exhibit rupture for a single event at impact velocities of 10 m/sec [32.8 ft/sec], as described in SNL Table 6-63 (2007ap). Therefore, the qualitative methodology used to evaluate waste package rupture probability for multiple impacts based on a 5-m/sec [16.4-ft/sec] impact velocity threshold is reasonable for use and would not significantly affect the results of TSPA analyses.

The NRC staff determines that DOE defined, in SNL Section 6.3.2.2.5 (2007ap), the maximum effective strain limit for the waste package rupture condition as 0.57 for uniaxial tension and 0.285 for biaxial tension. For realizations where the maximum effective strain was less than 0.285, DOE considered that rupture was not credible. When the maximum effective strain exceeded 0.285, the strain limit was multiplied by the triaxiality factor, resulting in an effective strain limit between 0.285 and 0.57. Finally, DOE evaluated the rupture condition on the basis of the newly computed strain limit. For some realizations, for which DOE concluded that the waste package did not rupture, the NRC staff noted that the computed maximum effective strains exceeded the effective strain limit [e.g., SNL Table 6-92 (2007ap)]. In response to the NRC staff's request for additional information, DOE stated in DOE Enclosure 3 (2009bq) that for all realizations with computed effective strain in the outer corrosion barrier greater than the uniaxial tensile strain limit of 0.57, as outlined in SNL Tables 6-60, 6-90, and 6-92 (2007ap), the stress state is compressive. Therefore, according to DOE, under these conditions the waste package rupture would not occur.

The NRC staff reviewed this information DOE submitted and notes exclusion of the waste package rupture is reasonable for these realizations of kinematic analyses. The NRC staff notes that a compressive state of stress would not lead to waste package rupture. Therefore, DOE's waste package damage results for the tensile tearing failure mode are reasonable for use as inputs for abstraction in its TSPA model.

Idealized State 2: Waste Package Structural Response Under Collapsed Drip Shield

Modeling Assumptions and Approach

DOE provided information on waste package structural response for the Idealized State 2 with a collapsed drip shield framework (SAR Section 2.3.4.5.4.3.2). DOE assessed deformations and stresses in the outer corrosion barrier of a TAD waste package loaded by a collapsed drip shield and the accumulated rubble. Outer corrosion barriers that were 17 and 23 mm [0.67 and 0.91 in] thick with intact and degraded internals were assessed. DOE's model represents the intact internals by the inner vessel, the TAD canister, and the fuel baskets with plates inside the canister. DOE assigned properties of Type 316 stainless steel to all internal components. The internals, which are assumed to be completely degraded, were represented by a material that can be considered to be similar to a weakly cohesive soil with no significant strength. This material fills the interior volume of the outer corrosion barrier to limit volume change to 50 percent.

DOE performed numerical analyses to assess the waste package structural response under a collapsed drip shield using the FLAC3D finite element models (SAR Section 2.3.4.5.4.3.2). In these analyses the drip shield was not explicitly modeled and was represented by bulkhead flanges that contact the waste package after collapse of the drip shield framework. DOE conducted these quasi-static analyses by applying vertical static loads to the drip shield bulkheads. The vertical loads were monotonically increased until pressures ranging from 500 to 1,500 kPa [10,400 to 31,300 psf] were reached. DOE considered that the average vertical static pressure from lithophysal rockfall for a complete drift collapse exerted onto the drip shield is 127 kPa [2,652 psf] (SAR Table 2.3.4-35). For the drip shield average vertical loading demand of 127 kPa [2,652 psf] (SAR Section 2.3.4.5.4.3.2.1), the maximum quasi-static pressures applied to the waste package are equivalent to peak ground accelerations (PGAs) in the range of about 3 to 9 g ("g" is acceleration due to gravity). DOE monitored the structural deformations and the residual stresses induced in the outer corrosion barrier as a function of the average vertical pressure exerted on the outer corrosion barrier by the drip shield bulkhead flanges.

NRC Staff's Evaluation of Modeling Approach and Assumptions

Using guidance in YMRP Section 2.2.1.3.2, the NRC staff evaluated the modeling approach for Idealized State 2 that DOE employed to evaluate the waste package response under quasi-static loading under the collapsed drip shield. For characterizing waste package damage, DOE followed established industry practice for mechanical/structural performance assessment using finite element methods (Bathe, 1996aa). DOE used the modeling calculations and represented the waste package and the drip shield and their components' geometries, including geometry simplifications, appropriately.

Representing the drip shield by bulkhead flanges is reasonable because the contact between the collapsed drip shield and the waste package that may cause damage is likely to occur between the drip shield bulkhead flanges and the waste package outer corrosion barrier. For Idealized State 2 analyses, DOE assumed that drip shield components have zero contact angles (i.e., lie flat) on the waste package outer corrosion barrier when vertical loads are applied. However, DOE initially did not provide a basis to support that the drip shield components would have a zero contact angle with the waste package if the drip shield framework collapses. In addition, the initial information in the SAR did not address how uncertainties in contact angle that result from differential deformation of the drip shield (e.g., partial framework collapse) or tilting of the waste package (e.g., due to the waste package emplacement pallet degradation) could affect the analyses for waste package damage. The NRC staff also considered that localization of stress from angular impacts may affect the localization of tensile strain in the outer corrosion barrier and, thereby, increase the likelihood of puncture or rupture (e.g., SAR Section 2.3.4.5.4.4.2).

In response to the NRC staff's request for additional information that addressed these issues, DOE provided additional information in DOE Enclosure 1 (2009br) to demonstrate the adequacy of its modeling approach. DOE provided waste package damage estimates that bounded waste package damage for angular impacts of the drip shield onto the waste package outer corrosion barrier. DOE stated that a partially collapsed drip shield could result in angular contact between the waste package outer corrosion barrier and the drip shield bulkhead. According to DOE, a partially collapsed drip shield does not completely lose its load-bearing capacity. DOE stated that a modeling approach that allows the drip shield to fully collapse onto the waste package (i.e., a modeling approach that produces a zero contact angle between the waste package outer corrosion barrier and the drip shield bulkheads) would overestimate the total load transferred to the waste package and, therefore, would overestimate waste package damage.

The NRC staff reviewed this information and noted that the overall modeling approach DOE used is reasonable for use for the following two reasons. First, for the residual stress failure mode, although an angular impact of the drip shield onto the waste package outer corrosion barrier could result in localization of stresses, the stress concentration areas would also be reduced. This would reduce the waste package outer corrosion barrier surface area that may be susceptible to stress corrosion cracking. Second, for the tensile tearing failure mode, DOE's analyses that do not consider an angular impact of the drip shield may underestimate the tensile tearing stresses. However, tensile rupture of the outer corrosion barrier would not occur for Idealized State 2, even for nonzero contact angles. This conclusion is based on the results of independent studies the NRC staff performed (Ibarra, et al., 2007aa,ab; Pomerening, et al., 2007aa). These independent studies showed that given the high ductility of Alloy 22, the waste package outer corrosion barrier would not be breached for the loads considered in the Idealized State 2 and nonzero angular impact. Therefore, DOE's modeling approach is reasonable for use, because it would not significantly affect the results of TSPA calculations.

Residual Stress Failure Mode

For the residual stress failure mode, DOE calculated the total damaged area as the sum of areas of all outer corrosion barrier elements (including interior and exterior surfaces) in which a single residual stress threshold of 90 percent of the Alloy 22 yield stress is exceeded (SAR Section 2.3.4.5.4.3.2). The NRC staff's confirmatory evaluation of the residual stress threshold values to estimate the waste package damaged area is presented in TER Section 2.2.1.3.1.3.2.3. For the analyses with a 17- or 23-mm [0.67- or 0.91-in]-thick outer corrosion barrier with intact internals, DOE made the following observations:

- The damaged area was less than 0.025 percent of the total outer corrosion barrier surface area for average vertical pressure up to 1,200 kPa [25,062 psf]

- The maximum damaged area was approximately 0.3 percent or less of the total outer corrosion barrier surface area for the highest evaluated vertical pressure of 1,500 kPa [31,328 psf]

- For the analyses with degraded internals, the vertical pressure of about 660 and 1,000 kPa [13,784 and 20,885 psf] may lead to a fully damaged waste package for 17- and 23-mm [0.67- and 0.91-in]-thick outer corrosion barriers, respectively

- For vertical pressure of less than or equal to 350 kPa [7,309 psf], the waste package damaged area was less than 0.1 percent of the total area (SAR Figure 2.3.4-93)

NRC Staff's Evaluation of Residual Stress Failure Mode

The NRC staff reviewed the residual stress failure mode results for these analyses using YMRP Section 2.2.1.3.2. DOE followed established industry practices in performing these finite element analyses, incorporated reasonable simplifications and assumptions, and used appropriate loading conditions to characterize the waste package damage. The NRC staff further notes that the results for the waste package damage DOE presented are technically defensible and therefore reasonable for use as input to the TSPA calculation. DOE's results are consistent with earlier studies by the NRC staff related to deformation shapes, strain, and stresses (Ibarra, et al., 2007ab).

Tensile Tearing Failure Mode

For the tensile tearing failure mode, DOE provided information on the maximum stresses in the waste package outer corrosion barrier for three vertical pressure levels—486, 807, and 1,483 kPa [10,150, 16,854, and 30,972 psf]. According to this information, the maximum stresses in the outer corrosion barrier did not exceed 420.4 MPa (SAR Figures 2.3.4-91 and 2.3.4-92), which is below the Alloy 22 ultimate tensile strength of 786 MPa, as detailed in SNL Table 4-3 (2007ap). Therefore, DOE reasonably demonstrated that rupture of the waste package outer corrosion barrier would not occur at these three vertical load levels (SAR Section 2.3.4.5.4.3.2).

NRC Staff's Evaluation of Tensile Tearing Failure Mode

The NRC staff reviewed the tensile tearing failure mode results for three vertical pressure levels analyses using YMRP Section 2.2.1.3.2 and, on the basis of the reasonableness of DOE's assumptions related to material behavior, establishment of initial and boundary conditions for the abstraction models, and comparison with NRC staff's independent studies (Ibarra, et al., 2007ab), noted that the results are reasonable.

Collapsed Drip Shield Condition

For Idealized State 2, with a collapsed drip shield framework, DOE concluded that

- State 2 bounds the case with intact waste package internals [tensile strain calculations from dynamic loads due to rock rubble after drip shield plate failure (Idealized State 3, which is reviewed in the section to follow)

- State 1 bounds State 2 for the case with degraded internals (the kinematic analyses for TAD waste packages)

However, DOE initially did not present the model results for tensile strain of the waste package after drip shield collapse (SAR Section 2.3.4.5.4.4.1). In addition, DOE initially did not discuss how free interactions between the waste package and drip shield, or dynamic interactions with rock rubble, appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package. In response to the NRC staff's request for additional information (DOE, 2009bs), DOE provided additional information intended to demonstrate that its results were bounding. DOE discussed how the free interactions between the waste package and drip shield and dynamic interactions with rock rubble appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package.

DOE performed a quantitative comparison of the maximum effective plastic strain results of (i) the kinematic analyses for impacts between the waste package and the pallet with degraded internals and (ii) the quasi-static analyses for the waste package with degraded internals loaded by a collapsed drip shield. On the basis of this comparison, DOE concluded that the maximum effective plastic strains from the kinematic calculations of impacts between a waste package and a pallet with degraded internals were greater. Thus, DOE concluded that the results bounded the effective plastic strains for the waste package with degraded internals loaded by a collapsed drip shield. In addition, DOE performed a quantitative comparison of the maximum effective plastic strain results of the kinematic analyses for the waste package surrounded by rubble and the quasi-static analyses for the waste package with its intact internals loaded by a collapsed drip shield. DOE concluded that the effective plastic strains from the calculations for the waste package surrounded by rubble were greater and, therefore, bounded the effective plastic strains for the waste package with intact internals loaded by a collapsed drip shield.

NRC Staff's Evaluation of Collapsed Drip Shield Condition

The NRC staff reviewed this information and compared quantitative results of the effective plastic strain DOE provided. The effective plastic strain results of both kinematic analyses bound the results of quasi-static analyses. Therefore, DOE's technical bases are reasonable for use and demonstrate that free interactions between the waste package and drip shield and dynamic interactions with rock rubble appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package. Further, the NRC staff notes DOE appropriately represented waste package performance for the Idealized State 2 (waste package loaded by a collapsed drip shield framework) in the TSPA evaluation.

Idealized State 3: Waste Package Structural Response in Direct Contact with Rubble Modeling Assumptions and Approach

DOE provided information on waste package structural response for Idealized State 3 where the waste package is in direct contact with rock rubble (SAR Section 2.3.4.5.4.3.1). DOE considered the loads produced by the weight of the rock rubble and the amplification of these loads during vibratory ground motion. These loads may lead to waste package damage through stress corrosion cracking, or rupture and puncture of the outer corrosion barrier. To examine the waste package damage potential, DOE performed mechanical/structural analyses of the TAD waste package in direct contact with the rubble. Two waste package outer corrosion barrier thicknesses of 17 and 23 mm [0.67 and 0.91 in] with degraded internals were considered. The system was subjected to static loads and dynamic amplification from ground motions with peak ground velocity levels of 0.4, 1.05, 2.44, and 4.07 m/sec [1.31, 3.44, 8, and 13.35 ft/sec] (SAR Section 2.3.4.5.4.1). These peak ground velocity values correspond to the ground motions presented on the bounded hazard curve in SAR Figure 2.3.4-18.

DOE conducted two-dimensional seismic analysis of the waste package surrounded by rubble using the computer code UDEC. The UDEC model initially represented an intact emplacement drift containing a waste package and pallet resting on the invert. The drift was allowed to collapse onto the waste package. Once static equilibrium was established, the model was subjected to ground motions and equilibrium was reestablished. DOE used a complete drift collapse simulation similar to the one used to assess potential drip shield framework buckling and drip shield plate rupture (SAR Section 2.3.4.5.3.2.1). The results included residual tensile stresses and effective tensile strains in the outer corrosion barrier. General observations on the deformed shapes of the outer corrosion barrier were also provided. DOE did not include the inner vessel, the TAD canister, or the fuel baskets in the waste package representation and only considered the degraded state of the waste package internals. DOE represented the degraded internals as a material similar to a weak cohesive soil with no significant strength. DOE stated that, for the geometrical representation used, the results of the TAD waste package provided a reasonable estimate of damage for the codisposal waste package. Therefore, separate models were not developed for the TAD and codisposal waste packages.

DOE used a two-dimensional plane strain representation of the waste package and its components for dynamic analyses under rubble loads, as outlined in SNL p. 6-216 (2007ap). This simplification assumes that the waste package extends infinitely in the direction normal to the calculation plane and that the structural response of the waste package is not affected by its boundaries (i.e., waste package lids). In SNL Appendix D (2007ap), DOE compared results of two-dimensional and three-dimensional stress analyses, using uniform static loadings that are not representative of the dynamic loads associated with seismic events. Because of the higher rigidity of the waste package lid area, the NRC staff considered that the outer corrosion barrier area in the vicinity of the waste package lid potentially could be more susceptible to tensile tearing than an open cylinder. The NRC staff raised this as a question and submitted it as an RAI to DOE.

In response to the staff's RAI, DOE provided additional information in DOE (2009bt) to demonstrate that the use of a two-dimensional waste package representation in seismic analyses of the waste package surrounded by rubble did not underestimate waste package damage. DOE stated that the two-dimensional waste package representation had reduced stiffness because the waste package lids that provide additional structural support were not included. The two-dimensional waste package representation was chosen because it maximizes structural deformation of the outer corrosion barrier. Further, DOE stated that three-dimensional waste package analyses were performed to investigate the potential for failure of waste package lids and connections between the waste package lids and the waste package wall. DOE concluded that these analyses demonstrated that tensile rupture of the outer corrosion barrier would only occur when the outer corrosion barrier collapses due to the waste package wall buckling, as described in SNL Appendix D (2007ap). DOE stated that because the two-dimensional waste package representation underestimates the loading demands needed for an outer corrosion barrier collapse, DOE concluded this representation would not underestimate the potential for the waste package outer corrosion barrier tensile failure.

NRC Staff's Evaluation of Waste Package Performance Under Accumulated Rubble Using a Two-Dimensional Model

The NRC staff reviewed the analyses and noted that DOE provided appropriate technical bases to show that the two-dimensional waste package representation in seismic analyses of the waste package surrounded by rubble did not underestimate the waste package damage for the residual stress failure mode and the waste package puncture probability. This is based on the following. The waste package damage for the residual stress failure mode and waste package puncture probability are functions of waste package deformations. This is a conservative approach to analyzing the performance: because DOE used a two-dimensional waste package representation, the kinematic analyses of the waste package surrounded by rubble would overestimate the waste package deformations. As a result, the waste package damage for the residual stress failure mode and waste package puncture probability would also be overestimated.

The NRC staff reviewed the kinematic analyses of the waste package surrounded by rubble analyses and noted that this modeling approach may underestimate the tensile tearing stress in the waste package outer corrosion barrier near the waste package lid. However, tensile rupture of the outer corrosion barrier in these locations is not likely to occur for the Idealized State 3 loading scenario. This is based on the following. In Idealized State 3, DOE performed kinematic analyses for the same set of ground motion time histories used for the Idealized State 1 evaluation. However, in Idealized State 3, the waste package is surrounded by rubble and the dynamic impacts would be distributed over a larger waste package contact area than in Idealized State 1. Redistribution of impact loads would reduce the potential for high strain/stress concentration regions and subsequent reduction in waste package damage. The NRC staff independently determined that the largest distributed impact forces on the waste package in Idealized State 3 do not exceed the maximum forces evaluated in the kinematic analyses for Idealized State 1. Moreover, in Idealized State 1, DOE appropriately demonstrated that waste package rupture is not likely to occur. As a result, DOE's assertion that waste package rupture is not likely to occur in Idealized State 3 is reasonable. Thus, the modeling approach DOE used is reasonable for use in the performance assessment because the model does not underestimate waste package damage for TSPA abstractions.

Residual Stress Failure Mode

For the residual stress failure mode, DOE concluded that the damaged area was generally a small percentage of the total waste package surface area. For the residual stress threshold of 90 percent of the yield stress, the damaged area resulted in 0.2 percent of the total waste package outer corrosion barrier surface area. For the residual stress threshold of 105 percent of the yield stress, the damaged area was about 3 percent of the total outer corrosion barrier surface area. DOE stated that the increase in damaged area correlated with an increase in peak ground velocity levels and thinning of the outer corrosion barrier.

NRC Staff's Evaluation of Residual Stress Failure Mode

The NRC staff reviewed the results of the residual stress failure mode using YMRP Section 2.2.1.3.2. DOE's response to NRC staff's request for additional information (DOE, 2009bt) demonstrated that two-dimensional waste package representation maximizes structural deformation of the outer corrosion barrier. Because higher waste package deformations would lead to higher residual stresses in the waste package outer corrosion barrier, the results for the residual stress failure mode are reasonable because they do not underrepresent waste package damage. Therefore, DOE appropriately represented the waste package damage results for the residual stress failure mode as input to the TSPA code.

Tensile Tearing Failure Mode

For the tensile tearing failure mode, DOE concluded in SNL Section 6.5.1.4.1 (2007ap) that the probability of rupture for the TAD and codisposal waste packages surrounded by rubble for the 17- and 23-mm [0.67- and 0.91-in]-thick outer corrosion barrier with degraded internals is zero.

DOE's conclusion was based on the observation that, for all simulations, the maximum effective plastic strain was below the ultimate tensile strain of Alloy 22.

For this idealized state, in addition to rupture probability, DOE calculated puncture probability of the waste package outer corrosion barrier. DOE considered that a severely deformed outer corrosion barrier may be punctured by the sharp edges of fractured or partially degraded internal components. DOE calculated a potential for puncture of the outer corrosion barrier. The calculation considered the reduction in the final cross-sectional area of a severely deformed outer corrosion barrier, as identified in SNL Section 6.5.1.4.1 (2007ap). DOE assumed that the probability of outer corrosion barrier puncture is zero until deformation of the waste package outer corrosion barrier is such that the diameter is reduced by 10 cm [4 in], as outlined in SNL p. 6-234 (2007ap). According to DOE, the puncture of the waste package outer corrosion barrier increased with increase in peak ground velocity and with decrease in the outer corrosion barrier thickness. Reported rupture probability ranges were from 0.01 to 0.82 for the 23-mm [0.91-in]-thick outer corrosion barrier with degraded internals and from 0.05 to 1.00 for the 17-mm [0.67-in]-thick outer corrosion barrier.

NRC Staff's Evaluation of Tearing Failure Mode

DOE used a failure criterion (i.e., ultimate tensile strain to evaluate the waste package rupture occurrence) that is consistent with accepted industry practice and is widely used in the field of mechanical/structural engineering (American Society of Mechanical Engineers, 2001aa). In SNL Section 6.5.1.2.2 (2007ap), DOE assessed the effective plastic stresses and strains of the final waste package configuration after reestablishing equilibrium. However, in the SAR, DOE did not explain whether effective stresses and strains were assessed at intermediate steps during the dynamic loading simulations. Because of the reversal of dynamic loading during modeled seismic events, the NRC staff questioned whether the effective plastic stresses and strains of final waste package configurations, after reestablishment of equilibrium, are consistent with the maximum effective plastic stresses and strains that occur during dynamic simulations.

In response to the NRC staff's request for additional information, DOE provided additional information in DOE Enclosure 2 (2009br) to demonstrate that using stresses and strains computed at the end of dynamic analysis is appropriate and does not underestimate damage to the waste package. DOE stated that, in the dynamic analyses of the waste package surrounded by rubble, the code cumulatively computes the effective plastic strain, and the plastic strains increase during the analyses' time history. In addition, DOE stated that the effective plastic strain is larger than the effective strain for the analyses with strain reversals and loading/unloading transitions. Therefore, DOE concluded that the use of effective plastic strain value obtained at the end of dynamic analyses is appropriate to evaluate the waste package damage. DOE stated that this approach would not underestimate the waste package rupture probability. On the basis of the reasonableness of the modeling assumptions and initial and boundary conditions used in DOE's analyses, DOE's technical bases are reasonable for use. Specifically, the evaluation of the stresses and strains at the end of dynamic analysis would not underestimate waste package damage. Thus, the analyzed waste package damage results are reasonable for use for the tensile tearing failure mode.

Puncture Probability

In calculating the waste package puncture probability, DOE assumed that the probability of the waste package outer corrosion barrier puncture is zero until deformation reaches a preset value. A waste package diameter reduction of 10 cm [4 in] was selected as the preset limit, as identified in SNL p. 6-234 (2007ap). Use of this assumption implies that the cross-sectional area of the outer corrosion barrier must decrease by 11 percent before the probability of puncture exceeds zero. This percentage decrease can be calculated from the ratio of the design basis waste package outer diameter to the waste package outer diameter reduced by 10 cm [4 in]. For the highest peak ground velocity level used, DOE calculated the probability of the 17-mm [0.67-in]-thick outer corrosion barrier puncture to be 0.20. This implies that 20 percent of the waste packages would be punctured during a seismic event at a 4.07-m/sec [13.35-ft/sec] peak ground velocity level.

NRC Staff's Evaluation of Puncture Probability

On the basis of the information discussed previously, the NRC staff notes that if DOE assumed that the waste package puncture probability exceeds zero for any deformation of the outer corrosion barrier, the number of punctured waste packages during a seismic event at a 4.07-m/sec [13.35-ft/sec] peak ground velocity level would only increase by 2 percent. Moreover, because the annual probability of exceedance for a 4.07-m/s [13.35-ft/sec] peak ground velocity level is 10^{-8}, this difference in the waste package puncture probability during the postclosure period would reduce even further. Therefore, DOE's assumption for puncture probability is reasonable because it would not significantly affect TSPA results.

In SAR Section 2.3.4.5.4.3.1.2 DOE stated that for a residual stress threshold of 90 percent of the yield stress, the damage area resulted in 0.2 percent of the total waste package outer corrosion barrier surface area. For a residual stress threshold of 105 percent of the yield stress, the damaged area was 3 percent. The NRC staff noted inconsistencies between this information and that provided in SAR Figure 2.3.4-89 and sought clarification. DOE provided the following clarification in its response to NRC staff's RAI, as outlined in DOE Enclosure 4 (2009bq):

"The DOE agrees that the percentages cited in the fourth and fifth sentences in SAR Section 2.3.4.5.4.3.1.2 are inconsistent with SAR Figure 2.3.4-89. The DOE will correct the numerical values in the fourth and fifth sentences of SAR Section 2.3.4.5.4.3.1.2 to read as follows:

> If the residual stress threshold is 90% of the yield strength, the average damaged area is less than 1.2% of the total outer corrosion barrier surface area. If the RST is 105% of the yield strength, the average damaged area is less than 0.1% of the surface area.

These numerical values are consistent with SAR Figure 2.3.4-89 and with the data in Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion (SNL 2007ap, Table 6-163)."

Summary of NRC Staff's Evaluation of Waste Package Mechanical/Structural Performance

The NRC staff reviewed DOE's information related to the mechanical disruption of engineered barriers (MDEB) abstractions for the waste package in the DOE performance assessment. This review was performed using guidance in YMRP Section 2.2.1.3.2 taking into account the risk significance of the waste package in the context of the repository postclosure performance. The NRC staff notes the following.

- For Idealized State 1, where the drip shield is structurally stable, DOE concluded that (i) dynamic impacts of the waste package with the rest of the engineered barrier system may cause damage to the waste package from end-to-end impacts between waste packages and between waste package and pallet and (ii) the extent of waste package damage for TSPA abstractions is a function of the waste package type, the waste package internals state, peak ground velocity levels, and the outer corrosion barrier thickness. DOE's conclusions are consistent with the analyses presented in the SAR and other supporting documents, and DOE adequately represented waste package performance for Idealized State 1 in the TSPA analysis.

- For Idealized State 2, with a collapsed drip shield framework, DOE concluded that (i) for the case with intact waste package internals, the waste package damage estimated for the Idealized State 3 is bounding and (ii) for the case with degraded internals, the waste package damage estimated for the Idealized State 1 is bounding. DOE's conclusions are reasonable for use because DOE has demonstrated, by comparing the results for all three idealized states, that the extent of waste package damage for Idealized State 2 bounds the waste package damage for Idealized States 1 and 3.

- For Idealized State 3, where the waste package is in direct contact with rock rubble, DOE concluded that (i) a waste package with a 23-mm [0.91-in]-thick outer corrosion barrier and degraded internals will not be damaged under seismic events with peak ground velocities below 2.44 m/sec [8 ft/sec] and (ii) the waste package damage depends on the waste package outer corrosion barrier thickness and the peak ground velocity levels. These conclusions are consistent with the analyses presented in the SAR and other supporting documents. DOE's waste package performance is reasonable for use for Idealized State 3 in the TSPA abstractions.

The NRC staff further notes that the technical bases for TSPA waste package abstractions presented in the SAR are reasonable for use because

- DOE appropriately addressed uncertainties in the waste package conditions and the environmental effects on the waste package components

- For characterizing waste package damage, DOE followed established practice for mechanical/structural performance assessment

- DOE used appropriate seismic loading conditions that are consistent with the values presented on the bounded hazard curve

- To evaluate the waste package damage, DOE used failure criteria that are consistent with accepted industry practice and/or widely used criteria in the field of mechanical/structural engineering

- For calculating the residual stress and establishing tensile tearing failure modes, DOE used analytical/numerical methods that are appropriate for the types of analyses

In summary, DOE's technical bases are reasonable for use for the waste package abstractions and these bases adequately represented waste package performance in the TSPA abstractions.

2.2.1.3.2.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for mechanical disruption of engineered barriers is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the performance assessment.

2.2.1.3.2.5 References

American Society of Mechanical Engineers. 2001aa. *ASME Boiler and Pressure Vessel Code*. New York City, New York: American Society of Mechanical Engineers.

Anderson, J.G. and J.N. Brune. 1999aa. "Probabilistic Seismic Hazard Analyses Without the Ergodic Assumption." *Seismological Research Letters*. Vol. 70. pp. 19–28.

Bathe, K.-J. 1996aa. *Finite Element Procedures*. Upper Saddle River, New Jersey: Prentice-Hall, Inc.

Bommer, J.J., N.A. Abrahamson, F.O. Strasser, A. Pecker, P.-Y. Bard, H. Bungum, F. Cotton, D. Fäh, F. Sabetta, F. Scherbaum, and J. Struder. 2004aa. "The Challenge of Defining Upper Bounds on Earthquake Ground Motions." *Seismological Research Letters*. Vol. 75. pp. 82–95.

BSC. 2008bl. "Supplemental Earthquake Ground Motion Input for a Geologic Repository at Yucca Mountain, Nevada." MDL–MGR–MG–000007. Rev. 00. ACN 01, ACN 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005aj. "Peak Ground Velocities for Seismic Events at Yucca Mountain, Nevada." ANL–MGR–GS–000004. Rev. 00. ACN 01, ACN 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004al. "Drift Degradation Analysis." ANL–EBS–MD–000027. Rev. 03. ACN 001, ACN 002, ACN 003, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bj. "Technical Basis Document No. 14: Low Probability Seismic Events." Rev. 1. MOL 20000510.0175. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bp. "Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada." ANL–CRW–GS–000003. Rev. 00. MOL20000510.0175. DOC20040223.0007. Las Vegas Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bq. "Mechanical Assessment of the Drip Shield Subject to Vibratory Motion and Dynamic and Static Rock Loading." CAL–WIS–AC–000002. Rev. 00A. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Budnitz, R.J., G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris. 1997aa. NUREG/CR–6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts—Main Report." Vol. 1. Washington, DC: NRC.

Coraddini, M.L. 2003aa. "Board Comments on February 24, 2003 Panel Meeting on Seismic Issues." Letter (June 27) to Dr. Margaret S.Y. Chu, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: United States Nuclear Waste Technical Review Board.

CRWMS M&O. 1998aa. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada." WBS 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O. 1998ab. "Synthesis of Volcanism Studies for the Yucca Mountain Site Characterization Project." 3781MR1. MOL 19981207.0393. Las Vegas, Nevada: CRWMS M&O.

DOE. 2010aa. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.2.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 4." Letter (January 29) J.R. Williams to J.H. Sulima (NRC). ML100290670. Washington, DC: DOE, Office of Technical Management.

DOE. 2010ab. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.2.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 4." Letter (February 12) J.R. Williams to J.H. Sulima (NRC). ML100470767. Washington, DC: DOE, Office of Technical Management.

DOE. 2010ac. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 3." Letter (January 28) J.R. Williams to J.H. Sulima (NRC). ML100290132. Washington, DC: DOE, Office of Technical Management.

DOE. 2009aa. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2.2.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.2, Set 2." Letter (January 27) J.R. Williams to J.H. Sulima (NRC). ML090280281. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ab. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2." Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550099. Washington, DC: DOE, Office of Technical Management.

DOE. 2009aq. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 1.1.10, 1.2.2, 1.1.5.2, and 1.1.5.3), Safety Evaluation Report Vol. 2, Chapter 2.1.1.1, Set 1." Letter (January 12) J.R. Williams to C. Jacobs (NRC). ML090270750. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bl. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.1, 2.3.11, 2.4.3, and 2.4.4), Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML091680648. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bp. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4.5.3.3.2), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 1." Letter (March 6) J.R. Williams to J.H. Sulima (NRC). ML090680836. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bq. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 2." Letter (November 24) J.R. Williams to J.H. Sulima (NRC). ML093360253. Washington, DC: DOE, Office of Technical Management.

DOE. 2009br. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 2." Letter (December 11) J.R. Williams to J.H. Sulima (NRC). ML093480212. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bs. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 2." Letter (December 15) J.R. Williams to J.H. Sulima (NRC). ML093500116. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bt. "Yucca Mountain—Response to Request for Additional information Regarding License Application (Safety Analysis Report Section 2.3.4), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.2, Set 2." Letter (November 30) J.R. Williams to J.H. Sulima (NRC). ML093350040. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 1997aa. "Topical Report YMP/TR–002–NP: Methodology To Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain." Rev. 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Ferrill, D.A., J.A. Stamatakos, and D. Sims. 1999ab. "Normal Fault Corrugation: Implications for Growth and Seismicity of Active Normal Faults." *Journal of Structural Geology*. Vol. 21. pp.1,027–1,038.

Ferrill, D.A., J.A. Stamatakos, S.M. Jones, B. Rahe, H.L McKague, R. Martin, and A.P. Morris. 1996aa. "Quaternary Slip History of the Bare Mountain Fault (Nevada) From the Morphology and Distribution of Alluvial Fan Deposits." *Geology.* Vol. 24. pp. 559–562.

Gray, M.B., J.A. Stamatakos, D.A. Ferrill, and M.A. Evans. 2005aa. "Fault-Zone Deformation in Welded Tuffs at Yucca Mountain, Nevada, U.S.A." *Journal of Structural Geology.* Vol. 27. pp. 1,873–1,891.

Ibarra, L., T. Wilt, G. Ofoegbu, and A. Chowdhury. 2007aa. "Structural Performance of Drip Shield Subjected to Static and Dynamic Loading." San Antonio, Texas: CNWRA.

Ibarra, L., T. Wilt, G. Ofoegbu, R. Kazban, F. Ferrante, and A. Chowdhury. 2007ab. "Drip Shield—Waste Package Mechanical Interaction—Progress Report." ML072740136. San Antonio, Texas: CNWRA.

Kana, D.D., B.H.G. Brady, B.W. Vanznat, and P.K. Nair. 1991aa. NUREG/CR–5440, “Critical Assessment of Seismic and Geotechnical Literature Related to a High-Level Nuclear Waste Underground Repository.” Washington DC: NRC.

Keefer, W.R., J.W. Whitney, and E.M. Taylor. 2004aa. “Quaternary Paleoseismology and Stratigraphy of the Yucca Mountain Area, Nevada.” U.S. Geological Survey Professional Paper 1689. Denver, Colorado: U.S. Geological Survey.

Livermore Software Technology Corporation. 2003aa. *LS-DYNA Users Manual*. Ver. 970. Livermore, California: Livermore Software Technology Corporation.

McGuire, R.K., W.J. Silva, and C.J. Costantino. 2001aa. NUREG/CR–6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Rick-Consistent Ground Motion Spectra Guidelines.” Washington, DC: NRC.

Morris, A.P., D.A. Ferrill, D.W. Sims, N. Franklin, and D.J. Waiting. 2004aa. “Patterns of Fault Displacement and Strain at Yucca Mountain, Nevada.” *Journal of Structural Geology*. Vol. 26. pp. 1,707–1,725.

Morris, A.P., D.A. Ferrill, and D.B. Henderson. 1996aa. “Slip-Tendency Analysis and Fault Reactivation.” *Geology*. Vol. 24. pp. 275–278.

NRC. 2009ab. “Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63.” Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, “Integrated Issue Resolution Status Report.” Rev. 1. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. Washington, DC: NRC.

NRC. 2003ae. Regulatory Guide 3.73, “Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations.” Washington, DC: NRC.

NRC. 1999aa. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity.” Rev. 2. Washington, DC: NRC.

NRC. 1997ab. Regulatory Guide 1.165, “Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion.” Washington, DC: NRC.

NRC. 1996aa. NUREG–1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.” Washington, DC: NRC.

Ofoegbu, G., R. Fedors, C. Grossman, S. Hsiung, L. Ibarra, C. Manepally, J. Myers, M. Nataraja, O. Pensado, K. Smart, and D. Wyrick. 2007aa. “Summary of Current Understanding of Drift Degradation and Its Effects on Performance at a Potential Yucca Mountain Repository.” Rev. 1. CNWRA 2006-02. ML071030115. San Antonio, Texas: CNWRA.

Pomerening, D., L. Ibarra, K. Hricisak, T. Wilt, K.T., Chiang, R. Kazban, and A. Chowdhury. 2007aa. “Experimental Tests on Drip Shield—Waste Package Mechanical Interaction—Progress Report.” San Antonio, Texas: CNWRA.

Schneider, J.F., N.A. Abrahamson, and T.C. Hanks. 1996aa. “Ground Motion Modeling of Scenario Earthquakes at Yucca Mountain: Final Report for Activity 8.3.1.17.3.3.” MOL19980617.0477. Las Vegas, Nevada: Yucca Mountain Project.

SNL. 2008ab. “Features, Events, and Processes for the Total System Performance Assessment: Analyses.” ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. “Total System Performance Assessment Model/Analysis for the License Application.” MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ap. “Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion.” MDL–WIS–AC–000001. Rev. 00. ERD 01, ERD 02 Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ay. “Seismic Consequence Abstraction.” MDL–WIS–PA–000003. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

Spudich, P., W.B. Joyner, A.G. Lindh, D.M. Boore, B.M. Margaris, and J.B. Fletcher. 1999aa. “SEA99: A Revised Ground Motion Prediction Relation for Use in Extensional Tectonic Regimes.” *Bulletin of the Seismological Society of America*. Vol. 89, No. 5. pp. 1,156–1,170.

Stamatakos, J.A., D. Biswas, and M. Silver. 2007aa. “Supplemental Evaluation of Geophysical Information Used To Detect and Characterize Buried Volcanic Features in the Yucca Mountain Region.” San Antonio, Texas: CNWRA.

Stamatakos, J.A., D.A. Ferrill, and K.P. Spivey. 1998aa. “Paleomagnetic Constraints on the Tectonic Evolution of Bare Mountain, Nevada. *Geological Society of American Bulletin.* Vol. 100. pp. 1,530–1,546.

Stamatakos, J.A., C.B. Connor, and R.H. Martin. 1997ab. “Quaternary Basin Evolution and Basaltic Volcanism of Crater Flat, Nevada, From Detailed Ground Magnetic Surveys of the Little Cones.” *Journal of Geology*. Vol. 105. pp. 318–330.

Waiting, D.J., J.A. Stamatakos, D.A. Ferrill, D.W. Sims, A.P. Morris, P.S. Justus, and K.I. Abou-Bakr. 2003aa. “Methodologies for the Evaluation of Faulting at Yucca Mountain, Nevada.” Proceedings of the10th International High-Level Radioactive Waste Management Conference, Las Vegas, Nevada, March 30–April 2, 2003. La Grange Park, Illinois: American Nuclear Society. pp. 377–387.

Wernicke, B., J.L. Davis, R.A. Bennett, J.E. Normandeau, A.M. Friedrich, and N.A. Niemi. 2004aa. “Tectonic Implications of a Dense Continuous GPS Velocity Field at Yucca Mountain, Nevada.” *Journal of Geophysical Research.* Vol. 109, No. B12404. p. 13.

Youngs, R.R., W.J. Arabasz, R.E. Anderson, A.R. Ramelli, J.P. Ake, D.B. Slemmons, J.P. McCalpin, D.I. Doser, C.J. Fridrich, F.H. Swan III, A.M. Rogers, J.C. Yount, L.W. Anderson, K.D. Smith, R.L. Bruhn, P.L. Knuepfer, R.B. Smith, C.M. dePolo, D.W. O'Leary, K.J. Coppersmith, S.K. Pezzopane, D.P. Schwartz, J.W. Whitney, S.S. Olig, and G.R. Toro. 2003aa. "A Methodology for Probabilistic Fault Displacement Hazard Analyses (PFDHA)." *Earthquake Spectra*. Vol. 19, No. 1. pp. 191–219.

(Intentionally Left Blank)

CHAPTER 6

2.2.1.3.3 Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms

2.2.1.3.3.1 Introduction

This Technical Evaluation Report (TER) chapter addresses those features, events, and processes (FEPs) included in the U.S. Department of Energy's (DOE) abstraction[1] of the repository drift system that may alter the chemical composition and volume of water contacting the drip shield and waste package surfaces (NRC, 2005aa). DOE described this abstraction in Safety Analysis Report (SAR) Sections 2.3.5 and 2.3.7 (DOE, 2008ab). This TER chapter provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the abstraction of key FEPs that address the following topics: (i) the chemistry of water entering the drifts, (ii) the chemistry of water in the drifts, and (iii) the quantity of water in contact with the engineered barrier system.[2] These three abstraction topics provide input needed to model the features and performance of the engineered barrier system (e.g., drip shield and waste package) and their contributions to barrier functions. For example, in its SAR, DOE relied on corrosion tests that were conducted on waste package and drip shield materials under a range of geochemical environments. The range of aqueous testing environments was deduced from a range of potential starting water compositions (Topic i) and from knowledge of near-field and in-drift processes that alter these compositions (Topics ii and iii).

2.2.1.3.3.2 Evaluation Criteria

The NRC staff's review of model abstractions used in the DOE postclosure performance assessment, including those considered in this chapter for the quantity and chemistry of water contacting engineered barriers and waste forms, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

---

[1]As used in the Technical Evaluation Report (TER), the term "abstraction" refers to the representation of the essential components of a process model into a suitable form for use in a total system performance assessment. A model abstraction is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses. An abstracted model is a model that reproduces, or bounds, the essential elements of a more detailed process model and captures uncertainty and variability in what is often, but not always, a simplified or idealized form.

[2]The abstraction of key features, events, and processes (FEPs) that address thermal-hydrologic processes affecting seepage rates are reviewed in TER Section 2.2.1.3.6; those that address corrosion processes affecting the drip shield and waste packages are reviewed in TER Section 2.2.1.3.1; and those that address the quantity and chemistry of water inside breached waste packages and the invert are reviewed in TER Section 2.2.1.3.4. Also, the review of the rationale for key FEPs that DOE has excluded from these abstractions is covered in TER Section 2.2.1.2.1.

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Section 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms involves seismic and igneous activity. 10 CFR 63.342(a) and (b) address the assessment of the effects of seismic and igneous activity on repository performance, subject to the probability limits given in 10 CFR 63.342(a) and (b). Specific constraints on the seismic and igneous activity analyses are in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

In addition, for this model abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms, 10 CFR 63.342(c)(2) further provides that DOE may assess climate change after 10,000 years by using a constant-in-time specification of the mean and uncertainty distribution for repository-average deep percolation rate for the period from 10,000 to 1 million years. DOE elected to use this representation in its Safety Analysis Report (SAR). Thus, implementation of the specified representative percolation rate and its uncertainty distribution is reviewed for the post-10,000-year period.

The NRC staff's review of the SAR and supporting information follows the guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Sections 2.2.1.3.3, Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff's review used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of the quantity and chemistry of water contacting engineered barriers and waste forms important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.3.3 Technical Evaluation

2.2.1.3.3.3.1 Chemistry of Water Entering Drifts

The abstraction for the chemistry of water entering drifts uses site-specific and literature-derived information as inputs to DOE's near-field chemistry model, which simulates chemical interactions of minerals in the Yucca Mountain host rocks with pore waters that percolate downward toward the repository. The model calculates (i) a water–rock interaction parameter that is used to predict initial seepage water compositions important to drip shield and waste package corrosion; (ii) radionuclide solubility [key parameters are pH, ionic strength (I), and concentrations of chloride (Cl-), nitrate (NO_3), and fluoride (F^-)]; and (iii) the range of in-drift carbon dioxide partial pressures ($p CO_2$). Important processes related to developing these parameters are discussed later in this chapter under the heading "Conceptual Model."

DOE used the results of its near-field chemistry model as inputs to other process-level models and direct inputs to the Total System Performance Assessment (TSPA) model. Potential seepage water compositions are used by the in-drift chemical and physical environment abstraction (reviewed in TER Section 2.2.1.3.3.3.2) and the waste package and drip shield corrosion abstraction at the process level (reviewed in TER Section 2.2.1.3.1). The range of in-drift $p CO_2$ values the near-field chemistry model predicts is used to generate a lookup table in the TSPA model, which is sampled to provide inputs to the waste form degradation and mobilization abstraction (reviewed in TER Section 2.2.1.3.4). TER Section 2.2.1.3.6 evaluates the abstraction that addresses thermal-hydrologic processes affecting seepage rates, TER Section 2.2.1.3.1 evaluates the abstraction that addresses corrosion processes affecting the drip shield and waste packages, and TER Section 2.2.1.3.4 evaluates the abstraction that addresses the quantity and chemistry of water inside breached waste packages and the invert.

In Safety Analysis Report (SAR) Table 2.1-3 and DOE (2009an), the chemistry of water flowing into drifts was recognized as important to the barrier capability of the emplacement drift, one component of the engineered barrier system. Water chemistry is important to performance because some seepage waters can have compositions that affect the corrosion of engineered materials. Localized corrosion of the waste package may occur if seepage waters of appropriate chemistry contact the waste package when the waste package temperature exceeds 23.4 °C [74.1 °F] (i.e., the ambient drift temperature prior to waste emplacement). The U.S. Nuclear Regulatory Commission (NRC) staff's review in Technical Evaluation Report

(TER) Section 2.2.1.3.1 considers the conditions under which localized corrosion is not expected to occur. In the U.S. Department of Energy's (DOE) system of engineered barriers, titanium drip shields prevent seepage water from contacting the waste package. DOE predicts that the drip shields will maintain their capability until compromised by mechanical or corrosive failure. The NRC staff's review of SAR Section 2.3.6.8 notes (see TER Section 2.2.1.3.1.3.1) that appreciable fluoride in the seepage water is needed to chemically compromise (by corrosion) the integrity of the drip shield. At DOE's predicted fluoride concentrations for nondisruptive scenarios, the drip shields will not appreciably corrode and will remain intact, as barriers to seepage contacting the waste package, during the first 250,000 years following closure. In the DOE model, as long as the drip shields remain intact, only slow general corrosion of waste packages occurs and only diffusional release of radionuclides is possible. With intact drip shields, significant amounts of seepage water are unlikely to contact the waste packages during the first 40,000 years following closure, when temperatures and relative humidity values are expected to exceed the pre-waste-emplacement conditions. After 40,000 years seepage water chemistries are expected to return to pre-waste-emplacement compositions with dilute concentrations of dissolved components. The NRC staff's review notes (TER Section 2.2.1.3.4) that these conditions will not significantly affect the mobility of radionuclides released from the waste package into the invert.

Mechanical processes are the other means by which drip shield performance may be compromised. DOE excluded early drip shield failure due to partial or complete collapse of drifts due to thermal effects (FEP 2.1.07.02.0A) on the basis of "low consequence." The rationale for excluding this specific FEP is reviewed in TER Section 2.2.1.2.1.3.2. The DOE performance assessment and the NRC staff's review note (TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) that few drip shields suffer mechanical failure within 12,000 years after repository closure. This is important because DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure, as described in DOE Enclosure 11 (2009dg). Beyond 12,000 years after repository closure, DOE calculated that there is a low probability for conditions in the drift to support localized corrosion of the waste package, if the drip shield fails and allows seepage water to contact the waste package, even given a somewhat elevated temperature of the waste package. The DOE model calculated that both the pH and nitrate-to-chloride ratio of water that may contact the waste package will generally be too high to initiate localized corrosion beyond 12,000 years after repository closure.

Because of the limited potential for the chemistry of water entering the drifts to affect performance significantly, the NRC staff conducted a simplified review that focused on the fundamental aspects of this abstraction. This approach is consistent with Yucca Mountain Review Plan (YMRP) guidance for conducting a risk-informed, performance-based review. Thus, the NRC staff's review focused on (i) the conceptual model, (ii) the initial range of pore water chemistries, (iii) the range of seepage water chemistries the near-field chemistry model predicted, and (iv) abstraction and integration. The review presented in this chapter is organized around these major topics and presented within the context of the DOE performance assessment evaluation. An assessment of the chemistry of seepage water that may contact the waste package or enter the invert during the time period when conditions (e.g., temperature, pH, seepage water chemistry) in the drift are affected by heat generated from radioactive decay of the waste is provided in TER Section 2.2.1.3.3.3.2.

Conceptual Model

This section addresses the system description and model integration (focused on system description), and the data used for model justification. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR) Section 2.3.5.3 (and relevant references) to evaluate the conceptual model of the chemistry of water entering the drifts.

SAR Table 2.2-1 contains the features, events, and processes (FEPs) that the U.S. Department of Energy (DOE) believes are potentially relevant to the chemistry of water entering drifts. DOE evaluated and included the following FEPs in this abstraction: (i) Chemical characteristics of groundwater in the unsaturated zone (FEP 2.2.08.01.0B), (ii) Chemistry of water flowing into the drift (FEP 2.2.08.12.0A), and (iii) Chemical effects of magma and magmatic volatiles (FEP 1.2.04.04.0B). DOE evaluated and excluded (on the basis of low probability or low consequence) the following FEPs from this abstraction: (i) Hydrothermal activity (FEP 1.2.06.00A), (ii) Altered soil or surface water chemistry (FEP 1.4.06.01.0B), (iii) Rind (chemically altered zone) forms in the near-field (FEP 2.1.09.12.0A), and (iv) Re-dissolution of precipitates directs more corrosive fluids to waste packages (FEP 2.2.08.04.0A). The exclusion of these FEPs from this abstraction is reviewed in Technical Evaluation Report (TER) Section 2.2.1.2.1.3.2. Furthermore, DOE's pre-10,000-year treatment of FEPs in this abstraction continues unchanged beyond the 10,000-year postdisposal period through the period of geologic stability.

The DOE conceptual model describes the chemical evolution of water as it percolates vertically toward the repository drifts. In the model, the water flows through the Topopah Spring repository host rock, a homogeneous unit that is 200 m [656.2 ft] thick, with average rock and hydrologic properties derived from measurements from equivalent units in the Yucca Mountain vicinity. Pore waters percolating through the unsaturated zone are modeled as chemically evolving by dissolution of alkali feldspar, which makes up about 60 percent of the host rock. Because of alkali feldspar's abundance, DOE assumed its dissolution represented host rock dissolution processes. The rate of feldspar dissolution increases as pore waters encounter elevated host rock temperatures that result from heat generated from radioactive waste decay.

After pore waters flow through the Topopah Spring rock to a location above the repository, the model calculates a chemical composition by combining one of four initial pore water compositions with an amount of feldspar predicted to have dissolved, and assuming chemical equilibrium with the minerals calcite and amorphous silica. Cation exchange onto clays or zeolites is not considered explicitly. Gas phase carbon dioxide (CO_2) concentrations are controlled by contributions from the CO_2 present in the local aqueous phase, CO_2 released from the evaporation of water (containing dissolved CO_2), and the partial pressure of CO_2 ($10^{-3.5}$ atmospheres) in the atmosphere. Calcite precipitation and feldspar dissolution influence the aqueous phase concentration of CO_2. Temperature also has a strong effect on CO_2 because this gas partitions more strongly to the gas phase at elevated temperatures.

The NRC staff compared DOE's description of the near-field chemistry conceptual model with the NRC staff's understanding of primary mineral dissolution and secondary mineral precipitation processes that control the chemical evolution of pore water as it percolates downward through the Yucca Mountain natural system. The NRC staff notes that while the included FEPs are very broad in nature, in abstracting them, DOE included the important chemistry-affecting processes and provided adequate technical bases for their inclusion in the abstracted model.

In addition, DOE considered the seismic and igneous intrusive scenarios in the abstraction of seepage water chemistry. The conceptual model for seepage water chemistry in the seismic scenario is the same as for the nondisruptive scenario described previously. For the igneous intrusive scenario, in which basaltic magma similar in composition to dikes found in the Yucca Mountain area fills much of the drifts, DOE considered the composition of seepage waters contacting the waste to be consistent with water that has reacted with basalt [BSC Section 6.3.1.3.5(a), (2005ad)]. DOE considered three basalt-influenced water compositions from large fractured basalt reservoirs (SAR Tables 2.3.7-10 and 2.3.7-11).

On the basis of the review of this information and knowledge of likely basalt–water interactions, the NRC staff notes that the three basalt groundwaters have chemical compositions that span the range of variation that could potentially enter a breached waste package following an igneous event and are reasonable for use in model simulations. Additional discussion of basaltic magma and its influence on seepage water is in TER Section 2.2.1.3.10. DOE showed in BSC Sections 6.5(a) and 6.6(a) (2005ad) that in-package chemistry is relatively insensitive to incoming water composition, whether from seismic scenarios or the igneous intrusive scenario. Consequently, the chemistries affected by igneous events, seismically induced ground motion, or fault displacement are adequate for their intended use in the SAR.

Initial Range of Pore Water Chemistries

This section addresses the characterization and propagation of data uncertainty through the model abstraction, and support of the model abstraction output by objective comparisons. The U.S. Department of Energy (DOE) described input parameter development and parameter uncertainty in Safety Analysis Report (SAR) Sections 2.3.5.3.2.2.1 and 2.3.5.3.2.2.2. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in the SAR (and relevant references) to evaluate the model inputs for the chemistry of water entering the drift abstraction. This evaluation focused on evaluating the uncertainty in the range of initial pore water chemistry [especially pH, ionic strength (I), and concentrations of chloride (Cl) and nitrate (NO_3)] and carbon dioxide partial pressures (pCO_2). In SAR Section 2.3.5.5 DOE identified these parameters as important inputs to the abstractions that deal with drip shield and waste package corrosion.

DOE's near-field chemistry model considers four initial pore water compositions as inputs. DOE assumed these four compositions represent the range of pore water compositions expected for the entire Yucca Mountain repository. A multistep screening process, based on charge balance and partial pressure of carbon dioxide, was used to evaluate 90 pore water analyses from Yucca Mountain cores that DOE deemed to be sufficiently complete for use in the near-field chemistry model. Thirty-four pore water compositions were identified as meeting the charge balance criteria (± 10 percent) and as having been minimally affected by microbial alteration (thus suitable for further consideration). Statistical cluster analysis was performed on the 34 samples, and 4 distinct groupings, or clusters, of samples were identified. The sample with the composition closest to the centroid of each cluster was selected as representative of each cluster.[3]

[3]Clustering analysis is a standard method for finding clusters of data that are similar in some sense to one another. The members of a cluster are more like each other than they are like members of other clusters. The centroid represents the most typical case in a cluster.

The NRC staff evaluated the information provided in SAR Section 2.3.5.3.2.2.1 and DOE (2009ck). The 56 pore water compositions that the DOE screening process eliminated from consideration had a median chloride-to-nitrate ratio 5 times greater than the 34 previously mentioned samples. DOE attributed the high chloride-to-nitrate ratios of the screened-out samples to the loss of nitrate by microbial activity during sample storage.

The NRC staff evaluated the support for the criteria used to screen the initial pore water compositions used as inputs to the near-field chemistry model (DOE, 2009ck). The NRC staff notes that while microbial activity during storage could result in high chloride-to-nitrate ratios, no direct evidence of such activity was presented by DOE. As a result, the NRC staff investigated the risk significance of any uncertainty that may have been introduced by excluding samples from the data set DOE used. The NRC staff did this by comparing the range and uncertainty in pore water compositions represented by DOE's 34 included pore water compositions with the NRC staff understanding of the Yucca Mountain natural system obtained from independent analyses of unsaturated zone geochemical processes and field observations at Yucca Mountain (Pabalan, 2010aa). The independent analysis considered 156 pore water samples the U.S. Geological Survey (USGS) collected from the unsaturated zone of Yucca Mountain. These 156 samples did not meet all of DOE's screening criteria for inclusion in the near-field chemistry model. However, the NRC staff notes that the data for these samples are sufficiently complete for this analysis, because the samples were characterized for pH, ionic strength, and chloride-to-nitrate concentration ratio. Thirty-three samples were selected to represent the range and distribution of the 156 pore water composition data set. This sample set represents a larger spatial distribution than the DOE samples and also bounds and enlarges the range of composition and concentration that DOE's 34 pore water compositions covered. The NRC analysis used StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) thermodynamic software to simulate the evaporative evolution of pore seepage waters. The StreamAnalyzer software uses a different electrolyte solution thermodynamic model and a more comprehensive thermodynamic dataset than the EQ3/6 code DOE used. The results of the analysis indicated that evaporation of initially dilute pore waters forms brines with compositions that do not support localized corrosion of the waste package and general corrosion of the drip shield. On the basis of this analysis, the NRC staff notes that considering a range of pore water compositions broader than the range DOE used in its near-field chemistry abstraction does not significantly affect the performance of the drip shield and waste package. Consequently, the DOE range of initial pore water compositions represented by the 34 pore water samples, while not bounding, is reasonable for use in the DOE Total System Performance Assessment (TSPA) model.

The Range of Seepage Water Chemistries Predicted by the Near-Field Chemistry Model

This section addresses the data used for model justification, and the characterization and propagation of model uncertainty through the model abstraction. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR) Section 2.3.5.3 (and relevant references) to evaluate the implementation of the conceptual model of the chemistry of water entering the drifts. As explained in TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6, the NRC staff notes that the U.S. Department of Energy (DOE) conclusion that few drip shields will fail within 12,000 years after repository closure as a result of corrosion or mechanical failure is reasonable. For the period 12,000 to 40,000 years after repository closure, the NRC staff expects that heat generated from radioactive decay of the waste would continue to affect conditions in the repository (i.e., temperature, relative humidity, pH, and chemical composition of in-drift waters). TER Section 2.2.1.3.3.3.2 assesses the chemistry of seepage water during the time period 12,000 to 40,000 years after repository closure, when conditions in the repository are affected by heat generated from radioactive decay of the waste. After 40,000 years, conditions in the repository are expected to return to the pre-waste-emplacement state. Consequently, the repository environment is not expected to alter the chemical compositions of seepage waters the near-field chemistry model predicted would enter the drift. As a result, this review focuses on the range of chemistries the near-field chemistry model predicted after 40,000 years.

DOE used the near-field chemistry model to determine the potential chemical compositions of seepage waters entering the drifts during the thermal period and when pre-waste-emplacement conditions return. The model uses a decoupled approach where hydrological and thermal processes are calculated independently. Chemistry is loosely coupled to the thermal and hydrological processes through the dissolution of alkali feldspar. The chemical composition of potential seepage waters is calculated at a location above the repository and at the bottom of the model domain using the geochemical speciation and reaction path code EQ3/6.

The NRC staff reviewed the information provided in SAR Sections 2.3.5.3.3.3 and 2.3.5.3.3.5 (and relevant references) and DOE (2009ck) to evaluate the model support for the range of seepage water compositions the near-field chemistry model predicted. The NRC staff reviewed DOE's comparison of the range of pH, chloride-to-nitrate ratio, and ionic strength values the near-field chemistry model predicted with the calculated values of pH, chloride-to-nitrate ratio, and ionic strength for the 34 pore water samples included in its abstraction. The comparison showed that the range of compositions the near-field chemistry model predicted for 40,000 years and beyond, when the repository returns to pre-waste-emplacement conditions, are not significantly different than those of the 34 starting pore water samples DOE included in its near-field chemistry model. The NRC staff also compared the range of compositions the near-field chemistry model predicted with the 56 samples DOE screened out from its near-field chemistry model and also with the 33 pore water compositions that NRC staff selected to represent the 156 USGS Yucca Mountain unsaturated zone pore water samples. The results of this comparison show that the range of seepage water pH, chloride-to-nitrate ratio, and ionic strength values the DOE near-field chemistry model predicted encompasses most (greater than 92 percent) of the pore water pH, chloride-to-nitrate ratio, and ionic strength values represented by the 34 screened in, 56 screened out, and 33 representative USGS samples. On the basis of this analysis, the NRC staff notes that the range of seepage water pH, chloride-to-nitrate ratio, and ionic strength values the DOE near-field chemistry model predicted under pre-waste-emplacement conditions is adequate for its intended use in the TSPA analysis.

Abstraction and Integration

This section addresses system description and model integration (focused on integration). The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR) Section 2.3.5.3.4 (and relevant references) to evaluate model integration and abstraction into the Total System Performance Assessment (TSPA) model of the chemistry of water entering the drifts. None of the results from the abstraction of the chemistry of water entering the drifts are directly used in the TSPA model. The near-field chemistry model provides inputs to the Engineered Barrier System Physical and Chemical Environment Abstraction Model. Results from the Engineered Barrier System Physical and Chemical Environment Abstraction Model are abstracted into the TSPA model. Both the Engineered

Barrier System Physical and Chemical Environment Abstraction Model and the abstraction of results into the TSPA model are evaluated in TER Section 2.2.1.3.3.3.2.

The near-field chemistry model calculates the thermal field using the same modeling approach and assumptions as other unsaturated zone thermal-hydrologic models. The NRC staff has evaluated the information provided in SAR Section 2.3.5.3.3.2.6 and compared it with the multiscale thermal-hydrologic modeling (SAR Section 2.3.5.4.1) and the in-drift condensation modeling (SAR Section 2.3.5.4.2). The NRC staff notes that the modeling approach DOE used in the SAR is consistent with these related abstractions and in the assumptions, approach, and parameters used.

Summary of NRC Staff's Review of Chemistry of Water Entering Drifts

Because of the limited potential for the chemistry of water entering drifts to affect drip shield and waste package performance significantly and the extent of DOE's corrosion testing programs, the U.S. Nuclear Regulatory Commission (NRC) staff's review focused on the fundamental aspects of this abstraction. The NRC staff reviewed the description of the near-field environment, the assumptions incorporated in the near-field chemistry model abstraction, the range of initial pore water compositions, the range of predicted seepage water compositions, and integration with other model abstractions. The NRC staff notes that the pre-10,000-year treatment of features, events, and processes (FEPs) in this abstraction continue unchanged beyond the 10,000-year postdisposal period through the period of geologic stability (defined as 1 million years). Therefore, DOE's treatment of FEPs through the period of geologic stability is reasonable.

In particular, considering the risk to performance, DOE appropriately described (i) the range of input data to the near-field chemistry model, (ii) important processes such as feldspar dissolution, and (iii) integration and consistency with other related model abstractions. The NRC staff further notes that the range of ambient temperature information passed to other abstractions, as well as the DOE corrosion testing program [e.g., pH, ionic strength (I) and concentrations of chloride (Cl) and nitrate (NO_3)] and gas phase partial pressures of carbon dioxide (pCO_2)], is reasonable.

2.2.1.3.3.3.2 Chemistry of Water in the Drifts

The abstraction for the chemistry of water in the drifts receives information on input gas and water compositions from the near-field chemistry model. The main purpose of the U.S. Department of Energy (DOE) in-drift water chemistry abstraction is to predict the range of chemical compositions for seepage on the waste package or in the invert for a given set of temperature, relative humidity, and pCO_2 conditions. This abstraction is implemented in the Total System Performance Assessment (TSPA) model in the form of lookup tables. These lookup tables enable the TSPA model to provide the parameters (and their uncertainties) needed to represent the chemical environment for the corrosion of waste package surfaces and for radionuclide transport in the invert.

The in-drift chemistry abstraction is not used to provide input to drip shield corrosion modeling. Instead, DOE modeled general corrosion of the titanium drip shield using two corrosion rate values based on weight-loss data determined from long-term corrosion tests. The U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the DOE drip shield general corrosion model abstraction is discussed in TER Section 2.2.1.3.1.3.1.1. In addition, TER Section 2.2.1.3.6 reviews the abstraction for thermal-hydrologic processes affecting seepage rates; TER Section 2.2.1.3.1 reviews the abstraction for corrosion processes affecting the drip shield and waste packages; and TER Section 2.2.1.3.4 reviews the abstraction for the quantity and chemistry of water inside breached waste packages and the invert.

According to DOE's Safety Analysis Report (SAR) Table 2.1-3 and DOE (2009an), the chemistry of water in the drifts is important to the capability of the emplacement drift (a barrier in the engineered barrier system). For example, incorrect representation of the chemistry of the waters in the drift may influence the calculation of waste package corrosion and radionuclide transport in the invert and may thus lead to incorrect dose estimates. Key risk information considered in assessing this abstraction includes (i) input data to the In-Drift Precipitates/Salts Model; (ii) consideration of processes that strongly affect water chemistry, such as evaporation, condensation, and salt precipitation; and (iii) uncertainty propagation through the model abstraction.

The following specific NRC staff's insights from the DOE performance assessment are important in evaluating this abstraction:

- During much of the thermal period, to about 12,000 years after repository closure, the drip shield is expected to prevent seepage water from contacting the waste package surface and greatly reduce the possibility of localized corrosion of the waste package.

- With no seepage water contacting the waste package within 12,000 years after repository closure and relatively limited expected waste package corrosion, only diffusive, not advective, release of radionuclides from the waste package is considered possible.

- For the period following 12,000 years after repository closure, DOE calculated that there is a low probability for the repository environment (i.e., temperature, pH, and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package.

- After 40,000 years, the temperature and relative humidity in the drifts will have returned to pre-waste-emplacement (or near pre-waste-emplacement) conditions and seepage water chemistry will also have returned to pre-waste-emplacement compositions with dilute concentrations of dissolved components. As a result of the low temperature, high relative humidity, and dilute seepage water chemistry, localized corrosion is not an important contributor to waste package degradation after 40,000 years.

Through its FEP screening process, DOE excluded several processes that might have been important to the chemical evolution of water in the drifts, such as corrosion due to deliquescence (FEPs 2.1.09.28.0A and 2.1.09.28.0B). The rationale for excluding specific FEPs from the in-drift water chemistry abstraction is reviewed in TER Section 2.2.1.2.1.3.2.

The NRC staff's review focuses on evaluating (i) the conceptual model inclusion of important processes, (ii) data and model justification, (iii) data and model uncertainty, and (iv) model abstraction support.

Conceptual Model and Important Processes

This section addresses the system description and model integration. The NRC staff reviewed information that the U.S. Department of Energy (DOE) provided in Safety Analysis Report (SAR) Section 2.3.5.5 (and relevant references) and DOE (2009cv,cw) to evaluate the conceptual model used to characterize the in-drift chemical environment. This evaluation focused on

(i) features of the conceptual model and (ii) important processes, because these specific topics are important to the DOE abstraction(DOE, 2008ab; NRC, 2003aa).

SAR Table 2.2-1 contains the features, events, and processes (FEPs) that DOE believes are potentially relevant to the chemistry of water in the drifts. DOE evaluated and included the following FEPs in the in-drift water chemistry abstraction: (i) Chemical characteristics of water in drifts (FEP 2.1.09.01.0A); (ii) Reduction-oxidation potential in drifts (FEP 2.1.09.06.0B); (iii) Reaction kinetics in drifts (FEP 2.1.09.07.0B); and (iv) Thermal effects on chemistry and microbial activity in the engineered barrier system (FEP 2.1.11.08.0A). DOE evaluated and excluded, on the basis of low probability or low consequence, the following FEPs in this abstraction:

- Chemical effects of excavation and construction in the engineered barrier system (FEP 1.1.02.00.0A)
- Undesirable materials left (in the repository) (FEP 1.1.02.03.0A)
- Seismic-induced drift collapse alters in-drift chemistry (FEP 1.2.03.02.0E)
- Chemical properties and evolution of backfill (FEP 2.1.04.02.0A)
- Erosion or dissolution of backfill (FEP 2.1.04.03.0A)
- Chemical effects of rock reinforcement and cementitious materials in the engineered barrier system (FEP 2.1.06.01.0A)
- Chemical degradation of invert (FEP 2.1.06.05.0D)
- Chemical effects at engineered barrier system component interfaces (FEP 2.1.06.07.0A)
- Gas explosions in the engineered barrier system (FEP 2.1.12.08.0A)
- Radiolysis (FEP 2.1.13.01.0A)
- Complexation in the engineered barrier system (FEP 2.1.09.13.0A)
- Microbial activity in the engineered barrier system (FEP 2.1.10.01.0A)
- Gas generation (CO_2, CH_4, H_2S) from microbial activity (FEP 2.1.12.04.0A)
- Radiological mutation of microbes (FEP 2.1.13.03.0A)
- Chemical effects of excavation and construction in the near-field (FEP 2.2.01.01.0B)

The rationale for excluding these specific FEPs from the in-drift water chemistry abstraction is reviewed in TER Section 2.2.1.2.1.3.2. Furthermore, DOE’s pre-10,000-year treatment of FEPs in this abstraction continues unchanged beyond the 10,000-year postdisposal period through the period of geologic stability.

DOE explained its conceptual model for in-drift chemistry as follows: early in the postclosure period, drift wall temperatures higher than the boiling point of water will prevent seepage from occurring. After the drift wall temperatures fall below the boiling point of water and the rewetting process begins, seepage may occur, as local hydrologic conditions allow. Because waste package surface temperatures will still be elevated, seepage water falling on the drip shield, and on the waste package in the event of drip shield failure, will evaporate and concentrate. As waste package temperatures continue to decrease, relative humidity will increase to the point that wet conditions persist. Over time, further increases in relative humidity will suppress evaporation and result in progressively more dilute aqueous solutions on the waste package surface or in the invert.

DOE's model considers how the chemistry of seepage water will evolve after it enters the repository drifts. In its conceptual model, DOE considered the effects of seepage water evaporation, condensation, gas–water interaction, precipitation and dissolution of salts, and salt separation. DOE's conceptual model describes in general terms how each of these processes influenced the chemistry of in-drift water. For example, seepage evaporation will cause the most soluble components to concentrate in the aqueous phase and minerals to precipitate. With precipitation, the relative concentrations of components remaining in solution will change. Salt separation may occur when seepage water flows downward over the drip shield or waste package surface while evaporation is occurring. During this process, spatial separation of chemical components could occur, transporting the more soluble aqueous components (e.g., NO_3^-) and leaving behind as precipitates the less soluble constituents (e.g., Cl^-, as NaCl precipitate). Condensation of water, which would dilute the aqueous phase concentration, could occur when the in-drift relative humidity is high enough. To model the in-drift water chemistry, DOE used the In-Drift Precipitates/Salts Model, which is a process-level geochemical model that accounts for the effects of in-drift processes. The In-Drift Precipitates/Salts Model was implemented using the EQ3/6 geochemical code and a Pitzer thermodynamic database that was developed for use in the In-Drift Precipitates/Salts Model. The NRC staff notes that the DOE model considered the most risk- and performance-significant processes, on the basis of NRC independent understanding of processes affecting the evolution of in-drift water chemistry (e.g., Murphy, 1994aa; Browning, et al., 2004aa; Yang, et al., 2011aa).

In natural systems, the chemical evolution of evaporating water generally is controlled by the high solubility of chloride and nitrate salt minerals relative to the moderate solubility of calcium sulfate and the low solubility of calcium carbonate minerals—a mechanism referred to as a chemical divide (Hardie and Eugster, 1970aa). Thus, evaporation of initially dilute natural waters at the Earth's surface, such as in saline lakes, typically leads to the formation of one of three brine types, depending on the initial composition of the system: calcium chloride brine, alkaline carbonate brine, or high sulfate brine. DOE concluded the same brine types could occur within the drifts because in-drift brines would be produced by processes similar to those that occur at the Earth's surface.

DOE used several assumptions in its abstraction of in-drift water chemistry. For example, all aqueous and gas constituents are assumed to achieve and maintain local equilibrium. The NRC staff considers this assumption reasonable because the chemical reactions considered in the abstraction are fast relative to the modeling timeframe. Also, the seepage waters on the waste package surface are assumed to reach equilibrium with the relative humidity on the waste package surface. The NRC staff notes that this assumption is conservative because the temperature would be highest and the relative humidity would be lowest at the waste package surface, which would maximize seepage water evaporation and result in the highest brine concentration. In addition, DOE assumed that an aqueous solution is present for all temperature and relative humidity conditions once seepage onto a waste package occurs. The NRC staff considers this a reasonable and conservative assumption given that an aqueous phase is necessary for corrosion of engineered barriers. This assumption is also supported by laboratory experiments in which dryout was not observed at temperatures up to 190 °C [374 °F], particularly for brines with Na–K–Cl–NO_3 compositions (Rard, et al., 2006aa). DOE also assumed the chemical compositions of drift wall condensation and of condensation that penetrates a failed drip shield to be the same as seepage composition and to be benign. This assumption is reasonable because water that condenses on the drift wall or other surfaces likely will be dilute, and any increase in concentration due to chemical interaction with drift wall surfaces, in-drift gases, and dusts deposited on waste package and drip shield surfaces will be small relative to increases in concentration due to evaporation.

In the salt separation abstraction, DOE assumed that the solution that forms during the salt separation process is chloride rich. The NRC staff considers this reasonable and conservative because a chloride-rich solution is more corrosive to the waste package material than a chloride plus nitrate solution.

With the In-Drift Precipitates/Salts Model, DOE conducted a series of seepage evaporation/dilution analyses at discrete temperature, relative humidity, and pCO_2 values. The analyses used as input the water compositions derived from the near-field chemistry model using 11 water–rock interaction parameters for each of the 4 representative pore water compositions—a total of 44 water compositions. DOE selected 3 temperatures for the analyses—30, 70, and 100 °C [86, 158, and 212 °F]—to cover the temperature range of interest while minimizing interpolation errors. At each temperature, the 44 waters were evaporated at 3 pCO_2 values: 10^{-2}, 10^{-3}, and 10^{-4} bar; these pCO_2 values were selected on the basis of the results from the near-field chemistry model. In a second set of EQ3/6 simulations, the waters were diluted by a factor of 100. The oxygen partial pressure (pO_2) in all the simulations was set equal to the atmospheric value to represent oxidizing conditions in the drift. The NRC staff considers this assumption reasonable given that the ranges for expected conditions are consistent with those predicted by DOE's performance assessment model and NRC staff's independent modeling (e.g., Murphy, 1994aa; Browning, et al., 2004aa). NRC staff notes that the pCO_2 values and pore water composition selection are reasonable, as discussed in TER Section 2.2.1.3.3.3.1.

The seepage evaporation/dilution analysis results formed the basis for the DOE in-drift water chemistry abstraction, which was implemented in the Total System Performance Assessment (TSPA) code in the form of 396 lookup tables (simulation results for 4 representative pore waters × 11 water–rock interaction parameter values × 3 temperatures × 3 pCO_2 values). The lookup tables represented the range of chemical compositions that potentially could be generated by evaporation or dilution of drift wall seepage or condensation, or by waters imbibed into the invert. The lookup tables enabled the TSPA code to provide the following parameters and their uncertainties for a given set of temperature, relative humidity, and pCO_2 conditions: pH, ionic strength, Cl^- and NO_3^- concentrations, and the NO_3^-/Cl^- ratio. These parameters are used in the TSPA model to represent the chemical environment for the corrosion of waste package surfaces and for radionuclide transport in the invert.

To determine which set of lookup tables is used for the in-drift water composition (SAR Section 2.3.5.5.4.3), the TSPA model used the following four inputs: the starting water identity (Groups 1, 2, 3, or 4); the water–rock interaction parameter derived from the near-field chemistry model; the pCO_2, which was derived from the near-field chemistry model; and the waste package surface temperature derived from the multiscale thermal-hydrologic model.

The specific water composition in the table was selected on the basis of the relative humidity at the waste package surface, which in turn was derived from the multiscale thermal-hydrologic model. For water–rock interaction parameters, temperatures, and pCO_2 values that fell between the values listed in the lookup tables, DOE interpolated values from adjacent tables.

DOE indicated that brine compositions resulting from seepage evaporation are most sensitive to the degree of water–rock interaction and to pCO_2; temperature had a comparatively smaller effect. The degree of water–rock interaction (the amount of feldspar dissolved) had the greatest effect on pH. With increasing amounts of feldspar dissolved, all the waters DOE considered in the analysis evolved into carbonate-type brines because feldspar dissolution and secondary mineral precipitation consume calcium and magnesium ions and raise the pH and bicarbonate concentration. The observed relationship between degree of water–rock interaction and brine type is important because carbonate-type brines typically have chloride and nitrate concentrations that are not conducive to localized corrosion of the Alloy 22 waste package outer barrier material. DOE concluded that corrosive calcium and magnesium-chloride brines are not expected to form in the potential repository.

The NRC staff compared DOE's conceptual model with the NRC staff's understanding of the Yucca Mountain natural system and independent analysis of in-drift processes (Browning, et al., 2003aa; 2004aa). On the basis of this understanding and risk-informed independent analysis, the NRC staff notes that DOE incorporated the appropriate physical processes and couplings in its conceptual model for in-drift water chemistry. The included features, events, and processes (FEPs) and associated physical processes and couplings in the abstracted model are consistent with independent geochemical modeling and consider relevant processes and parameters.

The DOE model calculated that for time periods beyond 12,000 years after repository closure, there is a low probability for the repository environment (i.e., temperature, pH, and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package. The DOE model calculated that the pH and nitrate-to-chloride ion ratio in in-drift water will generally be too high to initiate localized corrosion in this time period. The NRC staff conducted independent analysis of in-drift water that may contact the waste package under the temperature and relative humidity conditions that may exist in the drift at 12,000 years after repository closure or later. The NRC analysis used StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) thermodynamic software to simulate the evaporative evolution of seepage waters, using as input the compositions of USGS pore water samples discussed in TER Section 2.2.1.3.3.3.1. The results of the analysis indicate that brines resulting from evaporation of initially dilute pore waters do not support localized corrosion of the waste package at in-drift temperature and relative humidity conditions 12,000 years after repository closure or later. On the basis of this independent analysis, the NRC staff notes that the DOE results that indicate a low probability for localized corrosion initiation beyond 12,000 years after repository closure are reasonable, because the repository environment (i.e., temperature, pH, and chemical composition of seepage waters) would not support the initiation of localized corrosion (TER Section 2.2.1.3.1.3.2.2).

Data and Model Justification

This section addresses the data used for model justification. The NRC staff reviewed information that DOE provided in Safety Analysis Report (SAR) Section 2.3.5.5 (and relevant references) and in DOE (2009cw) to evaluate the data and model justification used to characterize the in-drift chemical environment. This evaluation focused on (i) the thermodynamic database and (ii) support of the model by laboratory experiments and other corroborating sources, because these specific topics are important to the DOE abstraction (DOE, 2008ab; NRC, 2003aa).

As indicated in the preceding section, the In-Drift Precipitates/Salts Model was implemented using the EQ3/6 geochemical code and a Pitzer thermodynamic database. The parameters in the database were obtained from a variety of thermodynamic data and solubility measurements reported in the scientific literature and synthesized into an internally consistent data set. DOE evaluated the principal temperature-dependent Pitzer parameters in the synthesized data set for their ability to reproduce the original source information. The NRC staff evaluated the comparisons of measured data and model results in SNL Appendix I (2007ao) and notes that the In-Drift Precipitates/Salts Model adequately reproduces the data used to construct the thermodynamic database.

DOE also used several chemical data sets to support its parameter values and to build confidence in the In-Drift Precipitates/Salts Model. The data sets included (i) laboratory experiments designed to investigate the effects of evaporation on the chemical evolution of water compositions and environmental conditions relevant to the potential repository; (ii) a natural analog for evaporative concentration of seawater at the Morton Bahamas solar salt production facility on Great Inagua Island in the Bahamas; and (iii) compilations of solubility measurements in single, binary, and ternary salt systems from handbooks and published sources. DOE compared these data with results from the In-Drift Precipitates/Salts Model in SAR Section 2.3.5.5.3.3 and referenced documents. The NRC staff evaluated these comparisons of measured data and model results and verified that, in general, the parameters used in the Pitzer thermodynamic database are reasonably supported by laboratory experiments, natural analog research, process-level modeling, and scientific literature (e.g., Linke,1958aa,1965aa; McCaffrey, et al.,1987aa; Wolf, et al.,1989aa; Rosenberg, et al., 1999aa, 1999ab; Alai, et al., 2005aa). However, the NRC staff determined that some model simulation results, such as those for individual and multisalt solutions, are inconsistent with experimental data. For example, although DOE reported that In-Drift Precipitates/Salts Model results for single, binary, and ternary salt saturation points and deliquescence relative humidities are generally within 20 percent of literature values, several In-Drift Precipitates/Salts Model results differ by 20 percent or more compared with experimental data. Although DOE asserted that the comparisons between literature and calculated values are favorable for individual and multisalt systems, figures provided for the single and multisalt systems show a mismatch between some calculated and experimental values, including a lack of similar trending between data sets. Furthermore, figures from SNL (2007ao) not included in the SAR show greater uncertainties than those included in the SAR. These figures show that as the complexity of the system increases, the dissimilarity between the values calculated by the In-Drift Precipitates/Salts Model and literature data appears to increase. However, the NRC staff notes that model justification is reasonable on the basis of propagation of uncertainty throughout the in-drift water chemistry abstraction (see next section), and because localized corrosion is not considered a contributor to waste package degradation after 12,000 years, because the in-drift water chemistry will not support localized corrosion of the waste package.

The NRC staff also evaluated DOE's baseline data used to justify the in-drift water chemistry abstraction. The NRC staff notes that the thermal, hydrological, and geochemical values used by DOE are justified. For example, the abstraction uses temperature and pCO_2 values that are technically defensible and that account for uncertainties and variability in those parameters. DOE described how measured pore water chemistry was used, interpreted, and synthesized into the in-drift water chemistry abstraction. However, the NRC staff believes that DOE did not consider the full range of natural system characteristics when DOE established the initial and boundary conditions for the in-drift water chemistry model. In particular, the four starting water compositions DOE used do not appear to represent the uncertainty in Yucca Mountain pore water composition. Nevertheless, as discussed in TER Section 2.2.3.3.3.1, NRC staff notes, on the basis of its understanding of near-field processes and from the results of its independent analyses, that incorporating a wider range of starting water compositions in the in-drift water chemistry abstraction will not significantly affect repository performance. Thus, the DOE baseline data justify the in-drift water chemistry abstraction given their intended use in DOE's Total System Performance Assessment (TSPA) model.

Data and Model Uncertainty

This section addresses the characterization and propagation of data and model uncertainty through the model abstraction. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed information that the U.S. Department of Energy (DOE) provided in SAR Section 2.3.5.5 (and relevant references) and in DOE (2009cw) to evaluate the data and model uncertainties used to characterize the in-drift chemical environment. This evaluation focused on (i) inclusion of uncertainty propagation in input data and (ii) uncertainty propagation throughout the In-Drift Precipitates/Salts Model, because these specific topics are important to the DOE abstraction (DOE, 2008ab; NRC, 2003aa).

DOE identified that uncertainties in the In-Drift Precipitates/Salts Model result in uncertainties in the TSPA code values of pH, ionic strength, concentrations of Cl^- and NO_3^-, NO_3^-/Cl^- ratio, and deliquescence relative humidity. DOE evaluated these uncertainties using model–data comparisons. Uncertainty in pH was given particular consideration due to variances in methods of measuring pH (whether true activity of the hydrogen ion is taken into account) and because there is significant experimental error when measuring the pH of concentrated brines. For solutions with water activities approximately 0.75 or higher, pH uncertainty was determined indirectly through the uncertainty in total inorganic carbon concentration, which reasonably reflects uncertainty in pH for the near-neutral range. This carbon concentration was evaluated using data from evaporation experiments and on calcite or CO_2 solubility. For more concentrated solutions with lower water activities, pH uncertainty was estimated on the basis of comparisons of calculated and measured pH in concentrated solutions. DOE evaluated the uncertainty in ionic strength by comparing values calculated using the In-Drift Precipitates/Salts Model with those derived from evaporation experiments. Uncertainties in the Cl^- and NO_3^- concentrations and in the NO_3/Cl^- ratio were evaluated by comparing In-Drift Precipitates/Salts Model results with data from evaporation experiments and solubility measurements.
DOE assessed the uncertainty in deliquescence relative humidity by comparing In-Drift Precipitates/Salts Model results with deliquescence relative humidity values reported in the literature.

The NRC staff evaluated DOE's characterization and propagation of uncertainty in the in-drift water chemistry abstraction. The NRC staff notes that uncertainties in the DOE conceptual model are considered and are consistent with available laboratory experiments and chemical data in published literature (e.g., McCaffrey, et al., 1987aa; Alai, et al., 2005aa). Specifically, uncertainties in pH, ionic strength, deliquescence relative humidity, and ionic concentrations derived from the In-Drift Precipitates/Salts Model are reasonably supported by laboratory evaporation experiments, solubility data, and deliquescence relative humidity data.

Model Abstraction Support

This section addresses support for the model abstraction output by objective comparisons. The U.S. Nuclear Regulatory Commission NRC staff reviewed information that the U.S. Department of Energy (DOE) provided in Safety Analysis Report (SAR) Section 2.3.5.5 (and relevant references) to evaluate the support for the model abstraction used to characterize the in-drift chemical environment. This evaluation focused on (i) consistency with process-level modeling and (ii) consistency of output data ranges with independent data, because these specific topics are important to the DOE abstraction (DOE, 2008ab; NRC, 2003aa).

In SAR Section 2.3.5.5.4.2.2, DOE described the approach it used to build confidence in the in-drift water chemistry model abstraction. For example, DOE abstracted the range of in-drift water chemistry in the form of lookup tables at discrete temperature and pCO_2 values. DOE supported the abstraction approach by showing that the results derived by interpolation between lookup tables are within the stated model uncertainties for In-Drift Precipitates/Salts Model simulations at the actual temperature and pCO_2 conditions tested. DOE provided additional support to its in-drift water chemistry model abstraction in DOE (2009cv). The NRC staff evaluated the DOE information and notes that the Total System Performance Assessment (TSPA) abstraction of in-drift water chemistry is consistent with process-level modeling. The NRC staff verified that the in-drift water chemistry abstraction is based on the same assumptions and approximations that are reasonable for process-level models of closely analogous natural or experimental systems.

The NRC staff also verified that DOE used accepted and well-documented procedures to construct and test the numerical model that simulates the evolution of in-drift water chemistry. The thermodynamic database used in the In-Drift Precipitates/Salts Model was developed from a variety of literature sources and synthesized into an internally consistent data set, which was evaluated for its ability to reproduce the original source information. The In-Drift Precipitates/Salts Model was supported by comparison with laboratory and natural analog information. Further, the NRC staff compared the ranges of in-drift water chemistry (e.g., pH, Cl^- and NO_3^- concentrations, ionic strength) DOE tabulated in lookup tables with the ranges derived from an alternative modeling approach the NRC staff used in its Total-system Performance Assessment code (Leslie, et al., 2007aa). The NRC modeling approach used thermodynamic calculations to simulate the evaporation of seepage waters. The calculations were implemented using StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) and used as input chemical compositions of Yucca Mountain unsaturated zone pore water samples the USGS reported (Yang, et al., 2003aa, 1998aa, 1996aa). The ranges in pH, Cl^- and NO_3^- concentrations, and NO_3^-/Cl^- ratio derived from the NRC approach are consistent with those derived from the DOE in-drift water chemistry abstraction. Thus, the NRC staff notes that the DOE abstraction output is supported by objective comparisons.

Summary of NRC Staff's Review of Chemistry of Water in the Drifts

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the in-drift chemistry abstraction (including the description of the in-drift environment), the assumptions incorporated in the In-Drift Precipitates/Salts Model Abstraction, the Pitzer database for the In-Drift Precipitates/Salts Model, supporting data and experiments, and uncertainty propagation through the In-Drift Precipitates/Salts Model. The NRC staff notes that the pre-10,000-year treatment of features, events, and processes (FEPs) in this abstraction continue unchanged beyond the 10,000-year postdisposal period through the period of geologic stability (defined as 1 million years). Therefore, DOE's treatment of FEPs through the period of geologic stability is reasonable.

In particular, on the basis of risk insight information, DOE appropriately (i) described input data to the In-Drift Precipitates/Salts Model; (ii) considered important processes such as evaporation, condensation, and salt precipitation; and (iii) propagated uncertainty through the model abstraction. The NRC staff notes that, for pre-waste-emplacement temperatures, the range of chemistry tabulated in the Total System Performance Assessment (TSPA) lookup tables (i.e., relative humidity; pCO$_2$ conditions of pH, ionic strength, Cl^- and NO_3^- concentrations; and the NO_3^-/Cl^- ratio) reasonably represents the potential chemistry of water contacting the surface of waste packages and radionuclide transport in the invert. The NRC staff also notes, on the basis of its independent analysis, that in-drift water chemistry is unlikely to initiate waste package localized corrosion in the time following 12,000 years after repository closure.

2.2.1.3.3.3.3 Quantity of Water in Contact With the Engineered Barrier System

The purpose for abstracting the quantity of water in contact with the engineered barrier system is to (i) determine seepage flux[4] rates through and around breached or intact waste packages and the drip shield and (ii) provide an estimate for partitioning of radionuclides exiting the engineered barrier system between unsaturated zone fractures and in the rock matrix. The engineered barrier system flow abstraction receives seepage flux approaching the drift wall from the drift seepage abstraction (BSC, 2004aa), condensation on the drift walls from the In-drift Natural Convection and Condensation model abstraction (BSC, 2004aw), capillary wicking (imbibition flux) into the invert from the Multiscale Thermal-Hydrologic Model abstraction (BSC, 2005ah), and the size and evaluation of corrosion openings on the waste packages from the WAPDEG Corrosion model abstraction (BSC, 2004bs). Finally, TER Section 2.2.1.3.6 reviews the abstraction that addresses thermal-hydrologic processes affecting seepage, TER Section 2.2.1.3.1 reviews the abstraction that addresses corrosion processes affecting the drip shield and waste packages, and TER Section 2.2.1.3.4 reviews the abstraction that addresses the quantity and chemistry of water inside breached waste packages and the invert.

The review of this abstraction is important because missing, discontinuous, or misrepresented flow paths in the engineered barrier system, and misrepresentation of seepage through and capillary diversions around the breached drift and waste packages, may result in incorrect dose estimates. Key risk information used to assess this abstraction includes (i) seepage flux rate at the drift wall, (ii) timing of the failure of the drip shield, and (iii) fraction of the breached patch area on waste packages. As discussed in the previous paragraph, this key risk information, which affects the distribution and flux rate throughout the engineered barrier system, is computed outside of this flow abstraction and then passed into this flow abstraction. Furthermore, the relative magnitude of the imbibition flux from the host rock matrix into the invert, which is computed outside the engineered barrier system flow abstraction, and the flux into the invert (that, in turn influences the fraction of radionuclides released from the engineered barrier system into unsaturated fractures and the rock matrix), which is computed by the engineered barrier system flow abstraction, are the key risk information propagated to the unsaturated zone transport abstraction in the Total System Performance Assessment (TSPA) model.

---

[4]Flux is the amount of water associated with a flow path at a given point along a pathway, at a given time. Flux rate is the amount of water per unit time.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed DOE's model for engineered barrier system flow in Safety Analysis Report (SAR) Section 2.3.7.12 (and relevant references). The NRC staff's review focused on evaluating (i) the conceptual model for flow paths and flux splitting throughout the intact and failed engineered barrier system components under both nominal and disruptive events; (ii) model integration of the engineered barrier system flow abstraction with other abstractions in the TSPA model, as well as information exchanges between the engineered barrier system flow abstraction and other abstractions; (iii) support for model parameters by available experimental data, and propagation of data uncertainties within the engineered barrier system flow abstraction and into other abstractions in the TSPA code; and (iv) analysis of model uncertainties through alternative model abstractions.

Conceptual Model for the Engineered Barrier System Flow Paths and Flux Splitting

This section addresses the system description and model integration (focused on system description/conceptual model). U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR) Sections 2.3.7.12 and 2.2.1.2.1 (DOE, 2009av) (and relevant references) and in DOE (2009ab) to evaluate the conceptual model of the quantity of water in the engineered barrier system. This evaluation focused on (i) the continuity and integration of flow paths and (ii) impacts of intact and breached engineered barrier system components on the engineered barrier system flow in the nominal case and disruptive events, because these specific topics are important to the U.S. Department of Energy's (DOE) abstraction (DOE, 2008ab; NRC, 2003aa).

Safety Analysis Report (SAR) Table 2.2-1 contains the features, events, and processes (FEPs) that DOE believes are potentially relevant to the quantity of water in contact with the engineered barrier system. DOE evaluated and included the following FEPs in this abstraction: (i) Capillary effects (wicking) in engineered barrier system (FEP 2.1.08.06.0A) and (ii) Unsaturated flow in the engineered barrier system (FEP 2.1.08.07.0A). DOE evaluated and excluded, on the basis of low probability or low consequence, the following features, events, and processes (FEPs) from this abstraction: (i) Advection of liquids and solids through cracks in the waste package (FEP 2.1.03.10.0A), (ii) Advection of liquids and solids through cracks in the drip shield (FEP 2.1.03.10.0B), (iii) Saturated flow in the engineered barrier system (FEP 2.1.08.09.0A), and (iv) Condensation on underside of drip shield (FEP 2.1.08.14.0A). Note that the rationale for excluding these specific FEPs from this abstraction is reviewed in TER Section 2.2.1.2.1.3.2. Furthermore, the U.S. Department of Energy's (DOE) treatment of FEPs in the pre-10,000-year period in this abstraction continues unchanged beyond the 10,000-year postdisposal period through the period of geologic stability.

DOE's engineered barrier system flow abstraction is based on a mass-conserving, flux-splitting algorithm involving eight potential unsaturated flow pathways in the engineered barrier system. The flow pathways and fluxes along these pathways are labeled F1 through F8 in SAR Figure 2.3.7-8 (DOE, 2009av). The upper wall of the drift forms the top boundary (along the F1 flow path), and the bottom part of the invert forms the lower boundary (along the F8 flow path) in the engineered barrier system abstraction. The abstraction calculates time-variant flux rates along unsaturated flow pathways across the engineered barrier system for the nominal case and disruptive events.

DOE described the flow pathways and flux rates (F1–F8) in the engineered barrier system flow abstraction as follows (Figure 6-1): the F1 flow path accounts for the total dripping flux from a drift wall. The total dripping flux is the sum of the seepage flux arriving at the drift wall from above and the condensed water on drift walls (TER Section 2.2.1.3.6.3.4); these are direct

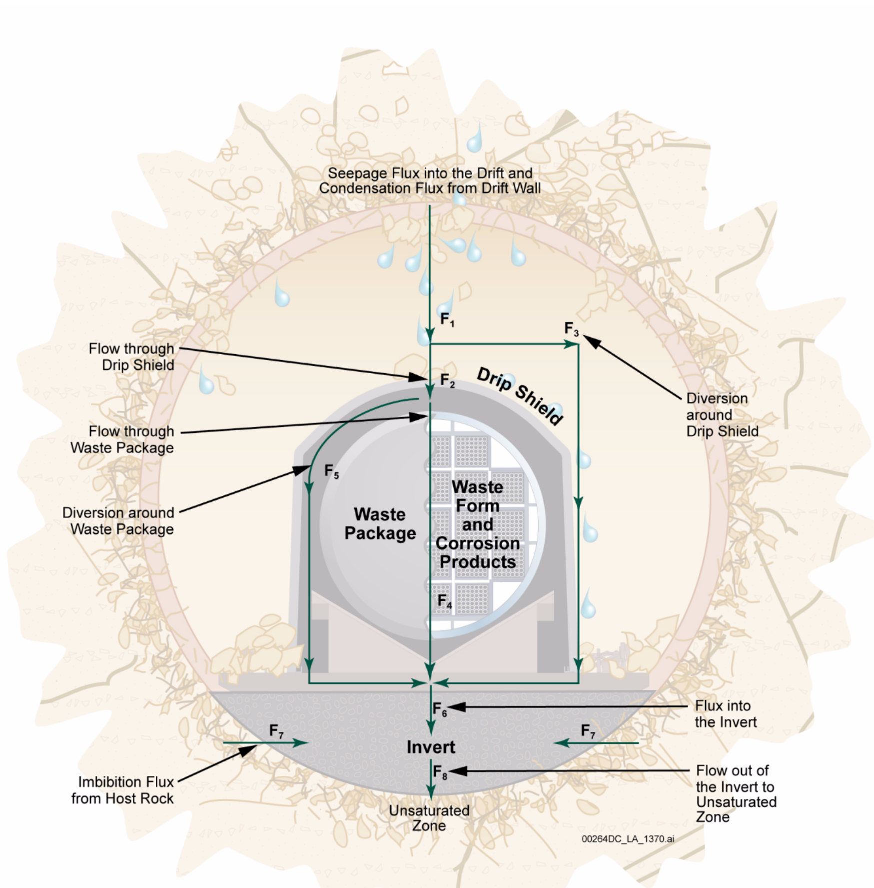

Figure 6-1. Potential Flow Pathways in the Engineered Barrier System (SAR Figure 2.3.7-8; DOE, 2008ab)

inputs to the engineered barrier system flow abstraction. The F2 flow path accounts for the flux through partially failed patches of the drip shield formed by general corrosion.

Localized corrosion of the drip shield is excluded from the performance assessment (TER Section 2.2.1.2.1.3). The F3 flow path accounts for the diversion of flux around the drip shield (computed as F3 = F1 – F2), which will drain directly into the invert. Although the diversion of flux around the drip shield is included in the construction of the engineered barrier system flow model, DOE did not implement the flux-splitting algorithm for drip shields in TSPA simulations, because the drip shields were modeled to be either all intact or failed, as described in SNL Section 6.1.1 (2007aj). The F4 flow path accounts for the flux through patches, formed as a result of general corrosion of the outer barrier of the waste packages. Localized corrosion of the outer barrier of waste packages is not considered important (TER Section 2.2.1.3.1.3.2.5). The F5 flow path accounts for the diversion of flux around waste packages (computed as F5 = F3 – F4), which will drain directly into the invert. The F6 flow path is the total flux entering the invert (computed as F6 = F4 + F3 + F5). The F7 flow path accounts for the imbibition flux from the host rock matrix into the invert and is a direct input to the engineered barrier system flow abstraction. The F8 flow path is the total flux from the invert to the unsaturated zone (computed as F8 = F6 + F7). Thus, the magnitude of fluxes in the engineered barrier system is determined by the flux rates at the drift wall, flow exchanges between the invert and the surrounding unsaturated fractured domain, and the size of corrosion patches on the drip shield and waste packages, which are externally calculated. The rest of the fluxes in the engineered barrier system are computed on the basis of the mass-conserving, flux splitting algorithm.

In addition to the nominal case, DOE addressed both the igneous intrusion and the seismic ground motion cases. The NRC staff confirmed that DOE showed, on the basis of TSPA simulation results, that these two disruptive modeling cases are the most significant contributors to the total dose for the 10,000- and 1-million-year simulations, as shown in SAR Figure 2.4-18 (DOE, 2009av). Because the contribution of the other modeling cases (including drip shield early failure, waste package early failure, seismic fault displacement, and volcanic eruption) to the mean annual dose for the 10,000- and 1-million-year simulations are at least an order magnitude smaller than the contributions to the mean annual dose by the igneous intrusion and the seismic ground motion modeling cases, the other modeling cases are not discussed in this section. For the igneous intrusion scenario, the drip shield and waste packages entirely lose their integrity instantaneously at the time of the intrusive event, and all seepage water approaching the drift wall flows through the waste package, as described in SNL Section 6.1.1 (2007aj). For the seismic ground motion scenario, after the drip shield fails, the water flow rate through a damaged waste package depends on the expected fraction of the waste package surface (which increases with time) that is breached by corrosion patches, as shown in SNL Figures 8.3-11(a) and 8.3-12(a) (2008ag) and DOE (2009an).

The NRC staff reviewed the model conceptualization, mass-balance equations, and underlying assumptions of the engineered barrier system flow model abstraction and other relevant abstractions with which the engineered barrier system flow model abstraction exchanges data and information to assess DOE's description of the engineered barrier system flow model abstraction and the underlying mass-conserving, flux-splitting algorithm. The NRC staff notes that DOE reasonably identified and described potential flow pathways and flow factors in estimating the quantity of water that could contact the engineered barrier system and waste forms on the basis of the mass-conserving, flux-splitting algorithm in the engineered barrier system flow abstraction. The engineered barrier system flow abstraction is reasonable because (i) the potential unsaturated flow pathways that DOE identified encompass all potential major flow pathways within the engineered barrier system, and between the engineered barrier system and the surrounding unsaturated fractured rocks; (ii) the impact of drip shield and waste package failures is addressed in the engineered barrier system flow abstraction by explicitly incorporating the failed or intact mode of the drip shield and incorporating externally computed temporal variations in the number of corrosion patches on waste packages; (iii) the impacts of the transient nature of dripping flux and flux exchange between the unsaturated zone and the invert are carried into the engineered barrier system flow abstraction through the F1 flow path and the F7 flow path, respectively; and (iv) loss of barrier capability of the engineered barrier system following an igneous event and gradual increase of the expected fraction of surface patches on waste packages, which account for increased water fluxes due to seismic ground motion (DOE, 2009an) are physically reasonable, and they are more risk significant as compared with the nominal case.

Model Integration and Information Flow

This section addresses the system description and model integration (focused on integration). U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR) Section 2.3.7.12 (and relevant references) to evaluate the model integration and information exchange with the other abstractions. This evaluation focused on (i) integration and continuity of flow components in the engineered barrier system flow abstraction and (ii) information on input to and output from the engineered barrier system flow abstraction, because these specific topics are important to the U.S. Department of Energy's (DOE) abstraction (DOE, 2008ab; NRC, 2003aa).

Input to the engineered barrier system flow abstraction includes: seepage flux into the drift from the drift seepage abstraction (BSC, 2004aa); condensation on the drift walls from the In-drift Natural Convection and Condensation Model abstraction (BSC, 2004aw), which makes up the F1 flow path; imbibition flux into the invert; the F7 flow path from the Multiscale Thermal-Hydrologic Model abstraction (BSC, 2005aa); and patch size and its evolution from the WAPDEG corrosion model (BSC, 2004bs), which is used for calculating the F4 flow path. The F4 flow path determines the seeping or nonseeping condition in the waste package. Information on the seeping or nonseeping condition is used in the engineered barrier system radionuclides and colloid abstraction for determining the rate constant for irreversible attachment of plutonium and americium onto mobile corrosion product colloids (DOE, 2009ay).

The U.S. Department of Energy's (DOE) SAR showed the engineered barrier system Unsaturated Zone Interface Model (SNL, 2007aj) uses water fluxes along the F6 and F7 flow paths to calculate advective flow rates of radionuclides and colloid suspensions to be used in the unsaturated zone transport abstraction. The F6 flow path determines the water flux rate for radionuclides and colloid suspensions from the invert into the unsaturated zone fractures in a seeping environment. In a nonseeping environment, advective flux from the invert to unsaturated zone fractures is zero unless drift wall condensation is greater than zero. The F7 flow path provides the water flux for advective transport of radionuclides and colloid suspensions from the invert into the unsaturated zone matrix. Imbibition along the F7 flow path could provide a small advective flux into the unsaturated zone matrix in both seeping and nonseeping environments (DOE, 2009am).

The NRC staff reviewed the model conceptualization, the mass-balance equations, and the underlying assumptions of the engineered barrier system flow model abstraction and other relevant abstractions with which the engineered barrier system flow model abstraction exchanges data and information to assess DOE's description of the integration of the engineered barrier system flow model abstraction and the information exchange with other abstractions in the Total System Performance Assessment (TSPA) model. DOE reasonably described model integration and information flow between the engineered barrier system flow abstraction and other abstractions in the TSPA model because DOE (i) identified and integrated input to (the F1 and F7 flow paths) and output from (the F6 and F8 flow paths) the engineered barrier system flow abstraction, (ii) computed fluxes internally across breached engineered barrier system components (the F2 and F4 flow paths) on the basis of time-variant information from other abstractions, and (iii) computed fluxes internally without needing direct input from other abstractions. DOE identified the upstream abstractions (drift seepage abstraction and In-drift Natural Convection and Condensation model), in-parallel abstractions (Multiscale Thermal-Hydrologic Model and WAPDEG corrosion model), and the downstream abstractions (engineered barrier system radionuclide transport abstraction), as well as the information exchanged among them through the F1, F2, F4, F6, F7, and F8 flow paths, under both seeping and nonseeping conditions.

Data Support and Uncertainties

This section addresses the data used for model justification, and the characterization and propagation of data uncertainty through the model abstraction. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in Safety Analysis Report (SAR)

Section 2.3.7.12 (and relevant references) to evaluate the supporting data and the characterization of uncertainties for the engineered barrier system flow abstraction. This evaluation focused on (i) experiments that address uncertainties associated with the number of drip points and flow rates; (ii) experimental data used to bound uncertainties associated with the engineered barrier system fluxes; and (iii) uncertainties propagated within the engineered barrier system flow abstractions and into the other abstractions in the TSPA code, because these specific topics are important to the U.S. Department of Energy (DOE) abstraction (DOE, 2008ab; NRC, 2003aa).

DOE relied on experimental data from the breached drip shield experiments, as outlined in SNL Section 6.3.2.3 (2007aj) and BSC (2003ag), to derive an equation to estimate flux through a breached drip shield (the F2 flow path). The equation is a function of the lateral spread angle of the rivulet flow on the drip shield, the number of corrosion breaches on the drip shield, the length of the breaches, and a sampled uncertainty factor. The uncertainty factor was bounded by DOE using data from the Breached Drip Shield experiment. DOE adopted the same equation to estimate the flux through a breached waste package (the F4 flow path). DOE identified that the only difference in implementing the equation for the drip shield and waste forms is that (i) the radius of the curvature of the waste package is less than that of the drip shield and (ii) the nominal patch size is smaller for a waste package than for the drip shield in the WAPDEG corrosion model (BSC, 2004bs). These differences affect the bounds for the uncertainty factor established for the drip shield and the waste package. DOE supported the abstraction for the flow through a breached drip shield and waste packages on the basis of data from the Breached Drip Shield experiment. However, in the TSPA code, the data support is used only for breaches on the waste package because the drip shields are modeled to be either all intact or failed, as detailed in SNL Section 6.1.1 (2007aj).

The NRC staff notes that uncertainties associated with the seepage flow arriving at the drift wall are propagated into the engineered barrier system abstraction through the F1 flow path. Uncertainties associated with unsaturated flow in the host rock matrix are propagated into the engineered barrier system flow abstraction through the F7 flow path. DOE bounded the uncertainty factor used, as described in SNL Sections 7.1.1.1 and 7.1.1.2 (2007aj), for the calculations of the F4 flow path on the basis of data from the breached drip shield experiments (BSC, 2003ag). DOE assumed a uniform distribution for the uncertainty factors due to lack of supporting data for any statistical distribution.

The NRC staff reviewed the Breached Drip Shield experiments to assess (i) the experiments designed to support the flux-splitting model conceptualization and (ii) the experimental data used to bound uncertainties in the engineered barrier system flow processes. The NRC staff notes that DOE used the data from the Breached Drip Shield experiments, in conjunction with simplified geometrical interpretations, as outlined in SNL Section 6.5.1.1.2 (2007aj), to characterize and bound the uncertainties associated with the F4 flow path (flux through a breached waste package). In particular, the NRC staff notes the following: (i) flow on the drip shield occurred as rivulets in the experiments, as expected from a physical standpoint, as a result of drips and splashes from a number of discrete drip points onto the drip shield; (ii) the experiments were run with the flow rate varying over 2 orders of magnitude {0.2–20 m^3/yr [52.8–5,283.4 gal/yr]}, covering a wide range of uncertainty in the flow rate; (iii) the experiments involved a wide range for drip points (1 to 90 drip points) directly above or away from patches to address uncertainties associated with drip locations (SAR Section 6.5.1.1.2.4); and (iv) the experiments provided the range for the splash angle and the effective drip shield, which were used for calculating uncertainties associated with the F4 flow path. During its review, the NRC staff calculated weighted seepage rates per waste package to be 0.01, 0.04, 0.05, and

0.08 m^3/yr [2.6, 10.6, 13.2, and 21.1 gal/yr] using the information in SAR Figure 2.3.3-47 (DOE, 2009av) for the mean seepage rate per waste package during the present-day, monsoon, glacial, and from 10,000 to 1 million years, respectively. These computed seepage rates are lower than the seepage rates used in the Breached Drip Shield experiments; however, NRC staff notes that the lower seepage rates in the Breached Drip Shield experiments are conservative from the perspective of radionuclide transport in the engineered barrier system because higher water flux rates would result in lower radionuclide mass concentrations.

Although upscaling and real-world repository conditions may introduce additional uncertainties, due to reasoning provided in (i) through (iv) in the previous paragraph, the NRC staff notes that the Breached Drip Shield experiments captured physical processes associated with the flow through the breached drip shield and waste packages. Hence, the use of the data from these experiments to bound uncertainties associated with the F4 flow path is reasonable. Because DOE implemented a mass-conserving flux-splitting algorithm in the engineered barrier system flow abstraction, and in light of the discussion in the previous section of this chapter on Model Integration and Information Flow (last paragraph), the NRC staff notes that (i) uncertainties associated with the drip flux and condensed water are propagated into the engineered barrier system flow abstraction through the F1 flow path, (ii) uncertainties associated with the number of patches on a breached drip shield and a waste package are propagated into the engineered barrier system flow abstraction through the F4 flow path, and (iii) uncertainties associated with flow conditions in the unsaturated zone around the invert are propagated into the engineered barrier system flow abstraction through the F7 flow path. Finally, the uncertainties associated with the engineered barrier system flow abstraction and data are propagated into the engineered barrier system radionuclide transport abstraction through the F6 flow path and the F8 flow path. Hence, data uncertainty is propagated within the engineered barrier system model abstraction and between the engineered barrier system model abstraction and other abstractions in the TSPA model.

Model Support and Uncertainties

This TER section addresses the characterization and propagation of model uncertainty through the model abstraction, and support of the model abstraction output by objective comparisons. U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided in SAR Section 2.3.7.12.3.5 (and relevant references) to evaluate model support and uncertainties for the engineered barrier system flow abstraction. This evaluation focused on (i) the alternative conceptualizations; (ii) the justification for the inclusion or exclusion of the alternative conceptualizations; and (iii) comparison of model output with the results from other process-level models, because these specific topics are important to the U.S. Department of Energy’s (DOE) abstraction (DOE, 2008ab; NRC, 2003aa).

DOE presented two alternative conceptualizations relevant to the engineered barrier system flow abstraction to characterize and propagate uncertainty through the model abstraction: the bathtub flow model and the dual-continuum invert flow model (SAR Section 2.3.7.12.3.5).

The engineered barrier system flow abstraction is based on a flux-splitting algorithm that assumes a nonponding (no water accumulation) condition in the engineered barrier system. DOE alternatively tested a ponding condition through a bathtub model that allows water retention and accumulation in the waste package before being released to the engineered barrier system. DOE identified that a flow-through (nonponding) model is bounding for the bathtub (ponding) model in calculations of concentration and mass releases of radionuclides from the engineered barrier system due to the delays in releases in the bathtub case, when

(i) radionuclides are solubility rate limited or dissolution rate limited and the inflow rate is time invariant or (ii) radionuclides are dissolution rate limited and there is a step change in the inflow rate. DOE identified that the flow-through model is not bounding for the bathtub model when the inflow rate increases, because the flow-through model (with an increased volumetric water flow rate) would result in lower mass concentrations than the bathtub model (with a fixed, completely mixed bathtub storage volume). However, the total mass releases (unlike the mass concentrations) passed from the engineered barrier system model abstraction to the unsaturated transport abstraction would be identical for the flow-through and bathtub models.

DOE discussed another alternative conceptualization in which the flow domain in the invert is characterized as a dual-continuum model, as opposed to the single-continuum model, in the engineered barrier system flow abstraction. In this alternative model, the flow domain is divided into intergranular and intragranular flow domains. As a result, the F8 flow path is redefined as the flux from the intragranular invert continuum to the unsaturated zone. DOE introduced an additional flow path, F9, that accounts for flux from the intergranular invert continuum to the unsaturated zone. DOE did not include this conceptualization in the TSPA model due to insufficient experimental data to validate diffusion when, in transport simulations, the water content is very low.

The NRC staff reviewed DOE's proposed alternative model conceptualizations to assess the rationale for their inclusion or exclusion in DOE's Total System Performance Assessment (TSPA) model. The NRC staff notes that the exclusion of the bathtub model from the TSPA code is reasonable because the flow-through model, as implemented in the TSPA code, is bounding for the bathtub model when flow rates are constant. On the other hand, DOE noted that when the inflow rate increases and radionuclides are solubility rate limited, the difference in the performance of the bathtub model and the flow-through model is not critical to performance. DOE's conclusion is reasonable because mass releases, rather than concentrations, of radionuclides are passed from the engineered barrier system to the unsaturated zone in the TSPA model and the mass of mobilized radionuclides (as a result of dissolution of waste forms) computed from the bathtub and flow-through models is identical in this case. The NRC staff further notes that the exclusion of the dual-continuum model from the TSPA code is reasonable because the spatial distribution of flow in the invert, or flow between and within the invert materials, is not significant in determining radionuclide releases from the engineered barrier system into the unsaturated fractures and unsaturated matrix in the TSPA model construction.

Further, DOE showed that the engineered barrier system flux-splitting algorithm tends to overestimate the fraction of drift flow that enters the breached mock-up drip shield (F2/F1) in the Breached Drip Shield experiments, as shown in SNL Table 6.5-2 and Figure D–12(2007aj. On the basis of experimental data, DOE estimated that the fraction of drift flow that entered the breached drip shield ranged from 0.013 to 0.275 with a median value of 0.049. DOE also used the results from the Breached Drip Shield experiments to estimate the fraction of drift flow that enters breached waste packages. Using the flux-splitting model, DOE calculated that when approximately 4 percent of the waste package surface area is breached by general corrosion, 10, 90, and 100 percent of the seepage flux approaching the (failed) drip shield from above enters into a breached waste package for the 5th, 50th, and 95th percentiles, respectively. In the TSPA model implementation, the average fraction of the breached waste package surface area in 1 million years is on the order of 10^{-3} for the nominal and disruptive modeling cases. DOE estimated that 0–11 percent (with a mean/median value of 5.5 percent) and 0–12 percent (with a mean/median value of 6 percent) of the seepage flux above the (failed) drip shield entered into a breached commercial spent nuclear fuel waste package for the nominal and seismic ground motion cases, respectively (DOE, 2009an). The NRC staff conducted similar calculations for a breached codisposal waste package for the nominal and seismic ground motion cases by following the same procedure and using information DOE presented in Safety Analysis Report (SAR) Figure 2.1-17 (DOE, 2009av). The NRC staff's calculations revealed that 0–12 percent (with a mean/median value of 6 percent) and 0–22 percent (with a mean/median value of 11 percent) of the seepage flux above the (failed) drip shield entered into a breached codisposal waste package for the nominal and seismic ground motion cases, respectively. The rest of the seepage flux was diverted around the breached waste package.

The NRC staff notes that on average 5–11 percent of seepage flux entering into a damaged commercial spent nuclear fuel or codisposal waste package in TSPA calculations is consistent with breach flux rates (which had a median value of 5 percent of the inflow rate) obtained from Breached Drip Shield experiments. Thus, DOE provided adequate modeling support using experimental data and TSPA model output.

Summary of NRC Staff's Review of Quantity of Water in Contact With the Engineered Barrier System

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the model conceptualization, mass-balance equations, the underlying assumptions of the engineered barrier system flow model abstraction and other relevant abstractions with which the engineered barrier system flow model abstraction exchanges data and information. The NRC staff also reviewed the Breached Drip Shield experiments DOE used to bound data uncertainties, and the alternative model conceptualizations DOE used to analyze model uncertainties. The NRC staff notes that the pre-10,000-year treatment of features, events, and processes (FEPs) in this abstraction continue unchanged beyond the 10,000-year postdisposal period through the period of geologic stability (defined as 1 million years). Therefore, DOE's treatment of FEPs through the period of geologic stability is reasonable.

The NRC staff notes that DOE adequately described (i) the engineered barrier system flow model abstraction involving failed and intact engineered barrier system components under the nominal case and disruptive events, (ii) input to and output from the engineered barrier system flow abstraction and the information exchange between the engineered barrier system flow abstraction and other abstractions in the Total System Performance Assessment (TSPA) code, (iii) data support for bounding uncertainties associated with fluxes through a breached drip shield and a waste package and their propagation across the engineered barrier system flow abstraction and into other abstractions in the TSPA code, and (iv) alternative model conceptualizations for analyzing model uncertainties.

Finally, the NRC staff notes that the igneous intrusion and seismic ground motion scenarios are the most significant contributors to the total dose for the 10,000- and 1-million-year simulations. In the Total System Performance Assessment (TSPA) code implementation, the fraction of seepage water at the end of 1 million years represents the maximum seepage flux that entered into damaged waste packages. TSPA calculations indicated that at the end of 1 million years, on average 5.5 percent of the seepage water approaching the (failed) drip shield entered into breached commercial spent nuclear fuel waste packages under nominal and seismic scenarios. Similarly, at the end of 1 million years, on average 6 and 11 percent of the seepage water approaching the (failed) drip shield entered into a breached codisposal waste package under nominal and seismic scenarios, respectively. Thus, on average, only up to 11 percent of the seepage flux above the (failed) drip shield would be available for the advective transport of radionuclides and colloids in the waste form and corrosion products domains in the engineered barrier system radionuclide transport abstraction. In this abstraction, a zero or nonzero value of the flux through a failed waste package determines the type of the transport mechanism for the radionuclides and colloids in the waste form and corrosion products domains (diffusive transport if the flux is zero; advective transport otherwise). Because, on average, not more than 11 percent of the seepage water can enter a waste package under any circumstances, the NRC staff notes that breached waste packages consistently divert a large fraction (more than 89 percent) of drift seepage. However, as shown in Figure 6-1, this diverted flux around the failed waste package enters into the invert and is used to calculate F6, which determines the advective transport of radionuclides and colloids in the invert domain of the engineered barrier system radionuclide transport abstraction. Therefore, this implementation is reasonable, because advective transport of radionuclides in the invert is controlled by water fluxes in the invert.

2.2.1.3.3.4 NRC Staff Conclusions

The NRC staff notes that the DOE description of this model abstraction for the quantity and chemistry of water contacting engineered barriers and waste forms is consistent with the guidance in the YMRP. The NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.3.5 References

Alai, M., M. Sutton, and S. Carroll. 2005aa. "Evaporative Evolution of a Na-Cl-NO_3-K-Ca-SO_4-Mg-Si Brine at 95 °C: Experiments and Modeling Relevant to Yucca Mountain, Nevada." *Geochemical Transactions*. Vol. 6. pp. 31–45.

Browning, L., R. Fedors, L. Yang, O. Pensado, R. Pabalan, C. Manepally, and B. Leslie. 2004aa. "Estimated Effects of Temperature-Relative Humidity Variations on the Composition of In-Drift Water in the Potential Nuclear Waste Repository at Yucca Mountain, Nevada." Proceedings of the Materials Research Society. Symposium Proceedings 824. J.M. Hanchar, S. Stroes-Gascoyne, and L. Browning, eds. Warrendale, Pennsylvania: Materials Research Society. pp. 417–424.

Browning, L., W. Murphy, C. Manepally, and R. Fedors. 2003aa. "Reactive Transport Model for the Ambient Unsaturated Hydrogeochemical System at Yucca Mountain, Nevada." *International Journal of Computers & Geosciences*. Vol. 29, No. 3. pp. 247–263.

BSC. 2005aa. "Analysis of Dust Deliquescence for FEP Screening." ANL–EBS–MD–000074. Rev. 01. AD 01, ACN 001, ACN 002. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005ad. "In-Package Chemistry Abstraction." ANL–EBS–MD–000037. Rev. 04. ACN 01, AD 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005ah. "Multiscale Thermohydrologic Model." ANL–EBS–MD–000049. Rev. 03. AD 01, AD 02, ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004aa. "Abstraction of Drift Seepage." MDL–NBS–HS–000019. Rev. 01. AD 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004aw. “In-Drift Natural Convection and Condensation.” MDL–EBS–MD–000019. Rev. 00. ACN 01, ACN 02, AD 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bs. “WAPDEG Analysis of Waste Package and Drip Shield Degradation.” ANL–EBS–PA–000001. Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2003ag. “Atlas Breached Waste Package and Drip Shield Experiments: Breached Drip Shield Test.” TDR–EBS–MD–000025. Rev. 00. Las Vegas, Nevada: Bechtel SACI Company, LLC.

DOE. 2009ab. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2.” Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550101. Washington, DC: DOE, Office of Technical Management.

DOE. 2009am. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.7, Set 1.” Letter (February 9) J.R. Williams to J.H. Sulima (NRC). ML090410352. Washington, DC: DOE, Office of Technical Management.

DOE. 2009an. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1.” Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Washington, DC: DOE, Office of Technical Management.

DOE. 2009av. DOE/RW–0573, “Yucca Mountain Repository License Application.” Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009ay. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report, Vol. 3, Chapter 2.2.1.3.4, Set 2.” Letter (May 12) J.R. Williams to J.H. Sulima (NRC). ML 091330282. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ck. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.3, Set 1.” Letter (April 30) J.R. Williams to J.H. Sulima (NRC). ML091210691. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cv. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.3), Set 1.” Letter (April 23) J.R. Williams to J.H. Sulima (NRC). ML091140343. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cw. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.3), Set 1.” Letter (April 16) J.R. Williams to J.H. Sulima (NRC). ML091100176. Washington, DC: DOE, Office of Technical Management.

DOE. 2009dg. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.6.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.1, Set 2." Letter (April 8) J.R. Williams to J.H. Sulima (NRC). ML090980537. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Gerbino, A. 2006aa. *A Guide for Using the OLI Analyzers*. Morris Plains, New Jersey: OLI Systems, Inc.

Hardie, L.A. and H.P. Eugster. 1970aa. "Evolution of Closed-Basin Brines." *Mineralogical Society of America Special Paper*. Vol. 3. pp. 273–290.

Leslie, B., C. Grossman, and J. Durham. 2007aa. "Total-system Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide." Rev. 1. ML072710060. San Antonio, Texas: CNWRA.

Linke, W.F. 1965aa. *Solubilities, Inorganic and Metal-Organic Compounds.* 4th Edition. Volume 2, K–Z. Washington, DC: American Chemical Society.

Linke, W.F. 1958aa. *Solubilities, Inorganic and Metal-Organic Compounds.* 4th Edition. Volume 1, A–Ir. Washington, DC: American Chemical Society.

McCaffrey, M.A., B. Lazar, and H.D. Holland. 1987aa. "The Evaporation Path of Seawater and the Coprecipitation of Br^- and K^+ With Halite." *Journal of Sedimentary Petrology*. Vol. 57. pp. 928–937.

Murphy, W.M. 1994aa. "Geochemical Models for Gas-Water-Rock Interactions in a Proposed Nuclear Waste Repository at Yucca Mountain, Nevada." Proceedings of Site Characterization and Model Validation Conference (Focus '93), Las Vegas, Nevada, September 26–29, 1993. La Grange, Illinois: American Nuclear Society. pp. 115–121.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

Pabalan, R.T. 2010aa. "Quantity and Chemistry of Water Contacting Engineered Barriers Integrated Subissue." Electronic Scientific Notebook 930E. San Antonio, Texas: CNWRA.

Rard, J.A., K.J. Staggs, S.D. Day, and S.A. Carroll. 2006aa. "Boiling Temperature and Reversed Deliquescence Relative Humidity Measurements for Mineral Assemblages in the $NaCl + NaNO_3 + KNO_3 + Ca(NO_3)_2 + H_2O$ System." *Journal of Solution Chemistry*. Vol. 35. pp. 1,187–1,215.

Rosenberg, N.D., K.G. Knauss, and M.J. Dibley. 1999aa. "Evaporation of J13 Water: Laboratory Experiments and Geochemical Modeling." UCRL–ID–134852. Livermore, California: Lawrence Livermore National Laboratory.

Rosenberg, N.D., K.G. Knauss, and M.J. Dibley. 1999ab. "Evaporation of Topopah Spring Tuff Pore Water." UCRL–ID–135765. Livermore, California: Lawrence Livermore National Laboratory.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aj. "EBS Radionuclide Transport Abstraction." ANL–WIS–PA–000001. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ao. "In-Drift Precipitates/Salts Model." ANL–EBS–MD–000045. Rev. 03. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

Wolf, M., O. Breitkopf, and R. Puk. 1989aa. "Solubility of Calcite in Different Electrolytes at Temperatures Between 10° and 60° C and at CO_2 Partial Pressures of About 1 kPa." *Chemical Geology*. Vol. 76. pp. 291–301.

Yang, L., R. Pabalan, P. Shukla, M. Juckett, X. He, K.T. Chiang, H. Gonzalez, and T. Ahn. 2011aa. "Corrosion of Alloy 22 Induced by Dust Deliquescence Brines." San Antonio, Texas: CNWRA.

Yang, I.C., Z.E. Peterman, and K.M. Scofield. 2003aa. "Chemical Analyses of Pore Water From Boreholes USW SD–6 and USW WT–24, Yucca Mountain, Nevada." *Journal of Contaminant Hydrology*. Vol. 1, No. 878. pp. 1–20.

Yang, I.C., P. Yu, G.W. Rattray, J.S. Ferarese, and J.N. Ryan. 1998aa. "Hydrochemical Investigations in Characterizing the Unsaturated Zone at Yucca Mountain, Nevada." USGS Water Resources Investigations Report 98-4132. Denver, Colorado: U.S. Geological Survey.

Yang, I.C., G.W. Rattray, and Y. Pei. 1996aa. "Interpretation of Chemical and Isotopic Data From Boreholes in the Unsaturated Zone at Yucca Mountain, Nevada." USGS Water Resources Investigation Report 96-4058. Denver, Colorado: U.S. Geological Survey.

CHAPTER 7

2.2.1.3.4 Radionuclide Release Rates and Solubility Limits

2.2.1.3.4.1 Introduction

This chapter addresses the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE) analytic models used in its Total System Performance Assessment (TSPA) computer program to simulate the processes that could result in water transport of radionuclides out of the engineered barrier system, including the waste package and the invert, and into the unsaturated zone (the rock mass directly below the repository horizon and above the water table). [As used in this Technical Evaluation Report (TER), the term "abstraction" refers to the representation of site characterization data; process-level models for features, events, and processes (FEPs); uncertainty and variability; and their overall integration (in a simplified manner) in the TSPA code.] These abstractions were described in Safety Analysis Report (SAR) Section 2.3.7 (DOE, 2009av) and in supporting documents, including DOE's responses to the NRC staff's requests for additional information. The objective of this review is to assess DOE's models for radionuclide release rates out of the engineered barrier system.

The engineered barrier system and the transport pathway within the drift (repository tunnel) are the initial barriers to radionuclide release. If a waste package is breached and water enters the waste package, the radionuclides contained in the package may be released from the engineered barrier system. The processes that could lead to radionuclide release are affected by the chemical characteristics of the water, which in turn are affected by the materials that interact with the water. The performance assessment analysis models radionuclide release rates from the engineered barrier system because these processes would significantly affect the timing and magnitude of transport for any radionuclide released from the repository.

DOE identified five models it considered important for abstracting radionuclide releases from the engineered barrier system. The five models DOE identified and the associated sections in this chapter that address them are

1. The in-package chemical and physical environment model (TER Section 2.2.1.3.4.3.1) used to establish the conditions under which waste forms degrade and radionuclides are mobilized

2. The waste form degradation model (TER Section 2.2.1.3.4.3.2) used to calculate the rate at which the waste form degrades and the radionuclides become available for release

3. The concentration limits model (TER Section 2.2.1.3.4.3.3) used to apply chemically based upper limits on dissolved concentrations of some radionuclides

4. The availability and effectiveness of colloids model (TER Section 2.2.1.3.4.3.4) used to calculate the stabilities and concentrations of various types of colloids (small suspended particles that may mobilize radionuclides in water)

5. The engineered barrier system radionuclide transport model (TER Section 2.2.1.3.4.3.5) used to simulate radionuclide transport from the waste form, through the waste package, and out of the engineered barrier system

The FEPs that are relevant to radionuclide release rates and solubility limits are listed in DOE's SAR Section 2.3.7.2 and Table 2.3.7-1. The NRC staff evaluates the rationales for excluding relevant FEPs from the performance assessment model in TER Section 2.2.1.2.1.3.2. In that section, the NRC staff notes that DOE's bases for the list of FEPs considered and excluded from the TSPA code analysis that are relevant to waste form behavior, solubility limits, colloidal transport, and radionuclide release rates are reasonable.

In addition, the NRC staff notes (TER Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4) that DOE's identification and screening of scenario classes considered all credible processes and events that could lead to radionuclide release. Evaluations of those FEPs included in the performance assessment are discussed under the five topical areas in this chapter.

This chapter relies on the following information as inputs: (i) design details of the waste package, waste form, and internal components of the waste package; (ii) context for consideration of the barrier capabilities of the waste package and the drift; (iii) information on corrosion and mechanical failure of the drip shield and waste package, which may allow water into the waste package; and (iv) information on the rate of delivery of water to the waste package surface and the chemical characteristics of water that may enter the waste package.

The output from the model of radionuclide release rates and solubility limits is used as input to the model for radionuclide transport in the unsaturated zone. The information the unsaturated zone model needs for calculating the movement of the radionuclides includes the rates and magnitudes of radionuclide release from the drift, including the characteristics of dissolved and colloidal species. Information from this model is also used for evaluating the barrier capability of the waste package interior, the waste form, and the drift below the waste package (e.g., the invert) and for supporting for the scenario analysis for the engineered barrier system.

2.2.1.3.4.2 Evaluation Criteria

The NRC staff's review of model abstractions used in the DOE postclosure performance assessment, including those considered in this chapter for radionuclide release rates and solubility limits, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction of radionuclide release rates and solubility limits involves seismic and igneous activity. 10 CFR 63.342(a) and 63.342(b) provide requirements for assessing the effects of seismic and igneous activity on repository performance, subject to the probability limits given in 63.342(a) and 63.342(b). Specific constraints on the seismic and igneous activity analyses are provided in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

The NRC staff's review of SAR and supporting information follows the guidance laid out in the Yucca Mountain Review Plan (YMRP) Section 2.2.1.3.4 (NRC, 2003aa), Radionuclide Release Rates and Solubility Limits, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria provide guidance for the NRC staff's evaluation of DOE's model abstraction of radionuclide release rates and solubility limits. These five criteria are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff's review used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of radionuclide release rates and solubility limits important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by

DOE and on the NRC staff's knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.4.3 Technical Evaluation

2.2.1.3.4.3.1 In-Package Chemical and Physical Environment

This section details the NRC staff's review of DOE's abstraction and the Total System Performance Assessment (TSPA) implementation of in-package chemistry, as described in Safety Analysis Report (SAR) Section 2.3.7.5 and references cited therein. The in-package chemistry model estimates the water chemistry inside the breached waste packages and generates abstractions for pH, ionic strength, and fluoride concentration. Water chemistry inside the waste package (especially pH and ionic strength) is important to repository performance because it controls waste form degradation, radionuclide solubilities, and the suspension stabilities of colloids.

The NRC staff's review focused on aspects of the in-package chemistry model considered significant to risk, including conceptual design and implementation, data inputs, model limitations, sensitivity to environmental conditions, and model abstraction and support. Primary data inputs to the in-package chemistry model include (i) the compositions, surface areas, and degradation rates of waste forms and metal components in the waste package; (ii) incoming water chemistries; and (iii) the thermodynamic data used to calculate the stabilities of dissolved, aqueous, and gas phase species in the waste package. DOE used the sensitivity of the model to variations in environmental conditions (e.g., liquid influx rate, pCO_2, and temperature) to determine the potential effects of disruptive events on model outputs.

Conceptual Design and Implementation

DOE's in-package chemistry conceptual model consists of a batch reactor system composed of water, oxygen, carbon dioxide, waste forms, and metal alloys. The batch reactor system is in equilibrium with atmospheric conditions, and reactants degrade in the presence of water according to a rate determined by the physical properties and exposed surface area of each reactant. During the reactions, secondary mineral phases and metal (hydr)oxide corrosion products precipitate and water changes in composition and mass. The model simulates two water ingress conditions: (i) vapor influx, under which water vapor (simulated as pure water) is assumed to condense and react with internal waste package components, and (ii) liquid influx, under which seepage or "dripping" water (simulated as typical groundwater or drift wall condensate) enters a breached waste package, reacts with internal components, and then exits by advection. Vapor influx is included in the model because water films generated by vapor influx promote radionuclide diffusion, which is simulated in the engineered barrier system using a diffusion model (TER Section 2.2.1.3.4.3.5).

DOE's model considers both commercial spent nuclear fuel (SNF) and codisposal waste packages. Commercial spent nuclear fuel waste packages contain 21 pressurized water reactor fuel assemblies (21-PWR). Codisposal waste packages contain two DOE multicanister overpacks and two defense high-level waste canisters (2-MCO/2-DHLW). DOE's model divides the waste packages into two domains: the waste form domain (Cell 1) and the corrosion products domain (Cell 2). Cell 1 of the codisposal waste packages is further divided into Cell 1a, represented by two high-level waste glass pour canisters (2–DHLW), and Cell 1b, represented by two multicanister overpack units containing N-Reactor fuel (2–MCO). Adsorption reactions are not simulated in the waste form cells, because the amount of iron corrosion products is low compared to Cell 2 (i.e., the corrosion products domain). Adsorption reactions within Cell 2 are simulated in the engineered barrier system flow and transport model (TER Section 2.2.1.3.4.3.5). DOE excluded other features, events, and processes (FEPs) that could potentially impact in-package chemistry, such as in-package criticality, oxide wedging, radiolysis, and microbial activity on the basis of low probability or low consequence. The NRC staff evaluates DOE's rationales for excluding these processes in Technical Evaluation Report (TER) Section 2.2.1.2.1.3.2.

DOE used the geochemical reaction path equilibrium modeling code EQ6 to simulate interaction of water and materials in commercial spent nuclear fuel Cell 1 and codisposal Cells 1a and 1b (referred to collectively as waste form cells). For vapor influx, water is added to the waste form cells as one of the reactants at a rate corresponding to the maximum diffusion rate of vapor through openings in a breached waste package. For liquid influx, the solid-centered flow-through option of the EQ6 code is used to simulate the flow of source water into and through a constant-volume, well-mixed batch reactor. Under the flow-through option, an amount of source water is added to the reactor displacing an equal amount of water already equilibrated with the solid phases in the reactor. The water in the reactor then mixes completely, and the water, solids, and gases within the reactor reequilibrate. Kinetically controlled reactants are also added to the reactor prior to equilibrium to capture the case where the residence time within the reactor is sufficiently short that equilibrium cannot be reached with slowly degrading constituents (e.g., metal degradation). The ratio of water to reactants, which depends on liquid influx rate, is treated as a variable in the EQ6 model. At high liquid influx rates, the ratio is such that the materials of the water package are in contact with a volume of water equal to that of the void space. This case is referred to as the "bathtub" model and has the highest ratio of water to waste package materials. At low liquid influx rates, the ratio is such that the volume of water in contact with waste package materials is less than the void space. In BSC Section 6.6.1[a] (2005ad), DOE examined the impact of varying the water-to-reactants ratio in a sensitivity analysis to evaluate the effects of differing flow conditions on in-package water chemistry. The sensitivity analysis indicated a negligible effect on pH but a distinct effect on ionic strength (i.e., as the ratio of water to reactants is decreased to simulate low flow conditions, the ionic strength of the solution increases).

NRC Staff's Review

The NRC staff evaluated the modeling approach and the representation of the commercial spent nuclear fuel and codisposal waste packages used in the in-package chemistry conceptual design. DOE's use of a thermodynamic equilibrium chemical approach of a flow-through, well-mixed batch reactor is reasonable because it represents in-package chemistry processes with sufficient flexibility to project the range of chemical conditions expected inside a breached waste package. Kinetics effects on chemical conditions inside the waste package are captured in the model by simulating a range of experimentally derived degradation rates for each waste form and metal component in the waste form cells. The effects of disruptive events on model outputs were evaluated through a sensitivity analysis to determine the effects of differing flow conditions on in-package water chemistry. The representation of commercial spent nuclear fuel and codisposal waste packages in the in-package chemistry model is consistent with waste package designs documented in SAR Section 1.5.2.

The NRC staff evaluated the EQ6 modeling code. On the basis of this evaluation, the mathematical representation of geochemical processes in the EQ6 modeling code is reasonable because it is consistent with generally accepted approaches for simulating geochemical interactions among fluids, gases, and solid materials. The EQ6 code is suitable for predicting the chemistry of in-package fluids because it addresses and simulates the geochemical processes that are important to in-package chemistry, including (i) interactions between codisposed waste, (ii) chemical effects of void space in the waste package, (iii) chemical characteristics of water in the waste package, (iv) oxidation-reduction potential in the waste package, (v) reaction kinetics in the waste package, and (vi) chemistry of water flowing into the waste package (SAR Table 2.3.7-1).

Waste Form and Metal Alloy Compositions, Surface Areas, and Degradation Rates

DOE derived input data for the in-package chemistry model from existing government design documents, standard reference material specification documents, and open literature information. The sources of input data to the in-package chemistry model were justified and documented in BSC Sections 4.1 and 4.1[a], Tables 4.1 and 4.1[a] (2005ad). The input data included the compositions, surface areas, and degradation rates of waste forms (e.g., pressurized water reactor or PWR fuel, high-level waste glass, and N-Reactor fuel) and material components of the waste form cells (e.g., stainless steels and aluminum alloys).

NRC Staff's Review

The NRC staff evaluated the input data DOE used to define the surface areas and compositions of waste forms and material components in the waste form cells by reviewing existing government documents and open literature information, as discussed and referenced in this paragraph. The NRC staff verified that for each solid reactant in the waste form cells, the surface area available to react was calculated from dimensions of the internal components of a 21-pressurized water reactor (PWR) or 2-multicanister overpack (MCO)/2-DOE high-level waste (DHLW) package (SNL, 2007bm,bn). Also, the NRC staff reviewed the composition of the pressurized water reactor fuel used for commercial spent nuclear fuel (Cell 1) that was based on an initial enrichment of 2 wt% to 5 wt% U-235 and a burnup of 0 to 50 GWd/MTU (gigaWatt-days per metric ton of uranium) (BSC, 2003af). These enrichments bound the typical PWR fuel available for disposal. The composition of N-Reactor fuel (Cell 1b) was taken from DOE (2000aa), which contains detailed information on multicanister overpack compositions and dimensions. The NRC staff reviewed the composition of high-level waste glass (Cell 1a) that was derived from qualified data on the composition of high-level waste glasses from the Savannah River laboratory (Allison, 2004aa). The compositions of material components in the waste form cells (i.e., stainless steels and aluminum alloys) were based on the American Society for Testing and Materials standards. On the basis of the NRC staff's evaluation of DOE's input data, the NRC staff notes that DOE used reasonable data to define the surface areas and compositions of each solid reactant in the waste form cells because they are consistent with published and qualified data and are appropriate for use in process-level in-package chemistry model simulations.

The NRC staff also reviewed the degradation rates DOE used. With the exception of the N-Reactor fuel, DOE selected degradation rates for waste forms and material components in the waste form cells on the basis of experimental measurements (BSC, 2004ae,ah,ai,ao). This approach is reasonable because DOE derived minimum, maximum, and basecase degradation rates for each solid reactant and captured the full range (i.e., uncertainty) of potential degradation rates in its analytic model. For the N-Reactor fuel, DOE assumed that the N-Reactor fuel degraded instantaneously upon waste package breach (SAR Section 2.3.7.8), which is conservative because this approach would maximize the effect of waste form degradation on in-package chemistry. The range of waste form and material degradation rates is reasonable for establishing initial and boundary conditions for in-package chemistry model simulations because these ranges are consistent with experimental measurements or are based on conservative assumptions.

Incoming Water Chemistry

DOE incorporated a range of Yucca Mountain pore water and basalt water chemistries in developing the in-package chemistry model abstractions. For seepage water input for the nominal and seismic scenarios, DOE selected four Yucca Mountain pore water compositions (SAR Table 2.3.7-9). DOE also included J–13 well water chemistry as a potentially relevant seepage water because its composition is generally representative of water compositions in the saturated and unsaturated zones in the vicinity of Yucca Mountain (Harrar, et al., 1990aa). For seepage water input for the igneous intrusion case, DOE selected three groundwaters from large, fractured basalt reservoirs (SAR Tables 2.3.7-10 and 2.3.7-11).

NRC Staff's Review

The NRC staff evaluated the chemistry of incoming waters used in the in-package chemistry model to simulate seepage water input. The NRC staff independently verified that the chemical compositions of the pore waters and the J–13 well water span the range of predominant water types found in the Topopah Spring welded tuff at Yucca Mountain, as described in SNL Section 6.6.5 (2007ak). DOE reasonably limited uncertainty in the initial chemistry of water entering the waste package by incorporating a range of Yucca Mountain pore water and basalt water chemistries in developing the in-package chemistry model abstractions. In BSC Sections 6.5(a) and 6.6(a) (2005ad) and in Safety Analysis Report (SAR) Figures 2.3.7-13 through 2.3.7-18, DOE showed limited sensitivity of in-package chemistry to the incoming water composition. Therefore, DOE used a reasonable range of incoming water chemistries to simulate seepage water input for developing the in-package chemistry model abstractions.

Thermodynamic Database

DOE used the thermodynamic database *data0.ymp.R5* to execute EQ6 simulations. This database allows for the calculation of mineral and gas solubilities, the chemical state of dissolved species, and the dissolution rates of solids. Uncertainty in the *data0.ymp.R5* thermodynamic database is implicit because it was constructed from the accumulation of a large number of experimental measurements or model estimations, each with its own associated uncertainty. To minimize this uncertainty, DOE examined experimental data and observations from natural analogs, as identified in BSC Section 7.4.3[a] (2005ad), and the results of sensitivity analyses, as detailed in BSC Section 6.6.11 (2005ad), to select appropriate secondary phase formation for use in process-level model simulations.

NRC Staff's Review

The thermodynamic database *data0.ymp.R5* is appropriate for the in-package chemistry model because it includes the elements that constitute the waste package, waste form, seepage, and gas compositions at the temperature expected in the drifts {25 °C [77 °F]} and the thermodynamic data on secondary mineral phases important to the in-package chemistry model (SNL, 2007at). The NRC staff also notes that DOE appropriately added thermodynamic data for several new mineral phases that could potentially affect model outputs. These data included nickel carbonate, nickel molybdate, and several uranium minerals that precipitate in UO_2 degradation experiments and occur in the vicinity of natural UO_2 ore deposits. The NRC staff reviewed the experimental data, evidence from natural analogs, and results of sensitivity analyses used to guide secondary phase selection for use in process-level model simulations. On the basis of this review, DOE's approach to limiting uncertainty in the EQ6 simulations associated with the *data0.ymp.R5* database is reasonable because appropriate secondary minerals were chosen for use in developing the in-package chemistry abstractions.

Model Limitations

DOE addressed model limitations associated with the accumulation of water inside the waste package and the evolution of material component surface area and void space due to corrosion product buildup by implementing the following assumptions in the process-level EQ6 simulations: (i) once a waste package is breached, the entire contents of the waste package are instantly exposed to oxygen, carbon dioxide, and water and (ii) secondary mineral formation and buildup inside the waste package do not reduce available void space in the waste package and do not reduce exposure of waste package internals to atmospheric gases and water (i.e., void volume and internal component surface areas are fixed and do not vary with time).

NRC Staff's Review

The NRC staff evaluated the assumptions DOE used to address model limitations. The NRC staff notes that instantly exposing the entire contents of the waste package to atmospheric gases and water upon breach is reasonable and conservative because it increases the potential for the model to predict enhanced radionuclide release. Reducing exposure of waste forms to gases and water by filling void spaces with degradation products enhances waste isolation by limiting the impact of waste form degradation on solution chemistry. Therefore, fixing void volume and waste package component surface areas is conservative because it results in faster waste form degradation, which enhances the potential for radionuclide release and transport. On the basis of these reviews, the assumptions DOE used to address model limitations are reasonable because they will not result in an underestimation of risk.

Environmental Conditions and Sensitivity Analyses

DOE developed the abstractions for in-package chemistry by analyzing the results of process-level model simulations applied over the following range of environmental conditions: (i) a pCO_2 range of 10^{-4} to $10^{-1.5}$ bars; (ii) a liquid influx rate of 0.1 to 1,000 L/yr [0.026 to 260 gal/yr]; (iii) a temperature range of 25 to 100 °C [77 to 212 °F]; and (iv) a relative humidity range for vapor influx of 95 to 100 percent. DOE performed sensitivity analyses to examine the effects of uncertain thermal-hydrologic-chemical input parameters on model outputs and approximate model uncertainty for propagation into the Total System Performance Assessment (TSPA) model. Parameters with significant effects on model outputs (e.g., pCO_2 for pH and liquid influx rate and relative humidity for ionic strength) were incorporated as independent variables in the model abstractions. Within the TSPA code, the values of these independent variables are provided by other TSPA submodels (e.g., the engineered barrier system thermal-hydrologic environment submodel provides relative humidity, the engineered barrier system chemical environment submodel provides pCO_2, and the engineered barrier system flow submodel provides liquid influx rate) (SNL, 2008ag). Parameters with smaller effects on model outputs (specifically, material degradation rates) were used to quantify model uncertainty for propagation to the TSPA model.

NRC Staff's Review

The NRC staff reviewed and evaluated the range of environmental conditions that was applied to process-level model simulations to develop the abstractions for in-package chemistry. The NRC staff notes that the applied range of pCO_2 will bound the range of pCO_2 that could potentially exist within emplacement drifts at Yucca Mountain. The applied range of liquid influx rate is appropriate because it bounds the range of flow rate used to characterize the uncertainty in seepage flux through a breached waste package (TER Section 2.2.1.3.3.3.3). Because liquid water, which is needed for the modeled reactions to take place, is precluded in DOE's abstraction from entering the drift at temperatures above 100 °C [212 °F], the NRC staff notes that DOE's model temperature range of 25 to 100 °C [77 to 212 °F] accounts for the temperature range expected inside breached waste packages. The NRC staff notes that the relative humidity range over which vapor influx conditions are simulated is reasonable because relative humidity less than 95 percent will result in few interconnected surface water films and negligible diffusion of radionuclides (SNL, 2007aj). On the basis of this review, the in-package chemistry model reasonably accounts for the range of environmental conditions that could reside inside breached waste packages under liquid and vapor influx conditions for the nominal and disruptive event scenario classes.

The NRC staff evaluated the sensitivity analyses used to examine the effects of uncertain thermal-hydrologic-chemical input parameters on model outputs. The NRC staff verified that pCO_2 had a significant effect on model outputs for pH, as detailed in BSC Section 6.6.3[a] (2005ad), and the liquid influx rate and relative humidity had significant effects on model outputs for ionic strength, as described in BSC Sections 6.6.4[a] and 6.5.2[a] (2005ad). The approach of incorporating these parameters as independent variables in the model abstractions (i.e., pCO_2 for pH and liquid influx rate and relative humidity for ionic strength) is reasonable because it ensures adequate integration and coupling of thermal-hydrologic-chemical processes in the model abstractions. On the basis of review of DOE's sensitivity analyses, the NRC staff notes that, after pCO_2, relative humidity, and liquid influx rate, the material degradation rates had the greatest effect on model outputs, as outlined in BSC Section 6.6.5[a] (2005ad). Therefore, using the results of material degradation rate sensitivity analyses to quantify model uncertainty for propagation to the TSPA model is reasonable.

Abstractions for pH

DOE's in-package chemistry abstractions for pH provide parameter distributions in the form of lookup tables for the TSPA code. Lower and upper pH limits for liquid and vapor influx in each waste form cell were quantified by simulated acid and base titrations over a range of pCO_2 and ionic strength (SAR Figures 2.3.7-19 to 2.3.7-21). Abstracted pH ranges were defined by secondary oxides, the presence of which limits the range of in-package pH through solubility reactions. The lower pH limit was set by dissolution of trevorite ($NiFe_2O_4$), which accumulates as the steel degrades. The upper pH limit was set by dissolution of schoepite, which precipitates as UO_2 fuel degrades and CO_2 reaches equilibrium conditions. To capture the uncertainty, the pH values at any given pCO_2 and ionic strength were assumed to be uniformly distributed between the pH limits established by the titration calculations. DOE supported estimated ranges for pH in the waste form cells by comparing predicted secondary mineral phases and pH ranges to observations from natural soils and groundwater, natural analogs, and/or laboratory experiments.

NRC Staff's Review

The NRC staff evaluated the modeling approach and information DOE used to generate and support the in-package chemistry abstractions for pH. The NRC staff verified that the solubility limit approach used to quantify lower and upper pH limits is consistent with accepted geochemical principles. The NRC staff evaluated information provided in DOE Enclosure 4 (2009ax) and notes that the choice of waste package design the model used (e.g., a 5-DHLW/DOE codisposal waste package containing five high-level waste glass containers versus the 2-MCO/2-DHLW codisposal waste package containing two high-level waste glass canisters) does not affect established pH limits, because the pH limits are based on buffering reactions that are not influenced by the total volumes and surface areas of material components in the waste form cells. In addition, on the basis of its review of information provided in BSC Sections 6.5[a], 6.6.3[a], and 6.6.5[a] (2005ad), the NRC staff notes that lower and upper pH limits defined for each waste form cell in the pH abstractions are reasonable because they are within the pH trends observed in time-dependent basecase EQ6 simulations at different incoming water chemistries and in sensitivity analyses at varying pCO_2 values and material degradation rates. On the basis of open literature reviews, the NRC staff notes that the phases DOE predicted to form and control pH in the waste form cells (i.e., trevorite and schoepite) are consistent with the phases reported as alteration products in steel corrosion and UO_2 degradation experiments, as well as in phases observed at natural analogs (Wang, et al., 2001aa; Da Cunha Belo, et al., 1998aa; BSC, 2004ah; Wronkiewicz, et al., 1996aa; Langmuir, 1997aa; Pearcy, et al., 1994aa).

DOE's abstracted pH ranges are consistent with the pH values measured in qualitatively similar soils and groundwaters and with pH ranges observed in UO_2 degradation experiments (Hem, 1995aa; Wronkiewicz, et al., 1996aa). On the basis of the NRC staff's evaluation, the solubility-controlling secondary oxide phases selected to quantify lower and upper pH limits in DOE's pH abstractions and the abstracted pH limits are reasonable.

Abstractions for Ionic Strength

As with pH, DOE's in-package chemistry abstractions for ionic strength provide parameter distributions in the form of lookup tables for the TSPA code. However, the manner by which ionic strength is abstracted differs under liquid and vapor influx conditions.

Under liquid influx conditions, DOE derived abstractions for ionic strength from a series of time-dependent EQ6 simulations at different liquid influx rates. DOE approximated uncertainty in ionic strength on the basis of variation in ionic strength observed in material degradation rate sensitivity analyses. At low liquid influx rates, the model generates high ionic strengths in the waste form cells (SAR Figures 2.3.7-22 to 2.3.7-24) because low flow rates provide sufficient residence time for the buildup in solution of waste form and metal alloy degradation products. DOE supported high ionic strength predictions in the waste form cells by comparing predicted ionic strengths to observations from natural groundwater and laboratory experiments.

For vapor influx conditions, DOE abstracted ionic strength as a function of relative humidity. At ionic strengths of 1 molal or less (relative humidity above ~98.5 percent), vapor influx was simulated using EQ6 and the B-dot equation to calculate activity coefficients to derive linear relationships between relative humidity and ionic strength (SAR Figure 2.3.7-25). When the ionic strength exceeded 1 molal (relative humidity at or below 98.5 percent), Pitzer calculations for simple salt solutions from the in-drift precipitates/salts model (SNL, 2007ao) were used to approximate the relationship between relative humidity and ionic strength, as described in BSC

Section 6.10.2.2[a] (2005ad). Uncertainty in ionic strength was derived from ionic strength variations observed in the Pitzer calculations.

NRC Staff's Review

For liquid influx, the NRC staff reviewed the modeling approach and technical support DOE used to generate ionic strength abstractions. The derived ionic strength ranges, from output of time-dependent EQ6 simulations using the *data0.ymp.R5* thermodynamic database, are appropriate because the ranges are calculated from a reasonably complete set of aqueous species at equilibrium. The sensitivity analyses results of liquid influx rate and material degradation rate on in-package ionic strength in BSC Sections 6.6.4[a] and 6.6.5[a] (2005ad) were reviewed. The variation in the liquid influx rate entering the waste package has a significant effect on ionic strength (i.e., ionic strength significantly increases as liquid influx rate decreases and significantly decreases as liquid influx rate increases). Therefore, deriving ionic strength as a function of liquid influx rate is reasonable because it provides a means for bounding in-package ionic strength over the entire range of flow conditions expected to enter a breached waste package. The variation in material degradation rates has a smaller effect on ionic strength. Therefore, approximating uncertainty in ionic strength as a function of material degradation rates is reasonable because it provides a means for bounding ionic strength on the basis of material components within the waste package. The NRC staff verified DOE's supporting information showing that the relationship between low liquid influx rate and high ionic strength is consistent with the evolution of deep groundwater brines in Canadian Shield granite (Appelo and Postma, 1994aa) and the results of UO_2 degradation experiments (Wronkiewicz, et al., 1996aa).

For vapor influx, the NRC staff reviewed the modeling approach used to generate the ionic strength abstractions. DOE's approach of deriving the ionic strength abstractions for vapor influx as a function of relative humidity is reasonable, because under vapor influx conditions, the water activity is controlled by relative humidity and ionic strength is strongly related to the water activity. At ionic strength values of 1 molal or less, deriving ionic strength from the output of EQ6 simulations using the *data0.ymp.R5* thermodynamic database is appropriate because the values are calculated from a reasonably complete set of aqueous species at equilibrium. However, for solutions with ionic strengths greater than 1 molal, deriving ionic strength from the output of EQ6 simulations using the B-dot activity coefficient is inappropriate because precipitates may form at high ionic strength resulting in large uncertainties in calculated pH and ionic strength. The NRC staff evaluated DOE's in-drift precipitates/salts model (SNL, 2007ao), which uses a Pitzer ion-interaction model to predict chemical conditions at high ionic strength. On the basis of this evaluation, the in-drift precipitates/salts model is reasonable for approximating the relationship between relative humidity and ionic strengths exceeding 1 molal.

Abstraction for Fluoride Concentration

DOE's fluoride abstraction provides maximum fluoride values for discrete ionic strength intervals for each waste form cell. Although high-level waste glass may contain some fluoride, the major source of fluoride in breached waste packages is from liquid influx (i.e., incoming water). Under vapor influx conditions (where water vapor, simulated as pure water, is assumed to condense inside the waste package), there is no significant source of fluoride in the waste form cells and maximum fluoride concentration is set to zero. Therefore, the fluoride abstraction is only applicable under liquid influx conditions. At high ionic strengths, maximum fluoride values were selected on the basis of relationships between fluoride concentration and ionic strength observed in material degradation rate sensitivity analyses at various incoming water compositions. At low ionic strength, maximum fluoride concentration was set to the maximum concentration observed in pore waters from the Topopah Spring welded tuff: 4.8 mg/L [0.00025 molal] (SNL, 2007ak).

NRC Staff's Review

The NRC staff evaluated the modeling approach and information DOE used to generate and support the in-package chemistry abstraction for fluoride. On the basis of its review, the NRC staff notes that the major source of fluoride in breached waste packages comes from liquid influx. At high ionic strengths, fluoride often concentrates as incoming water is consumed by degradation reactions, resulting in fluoride levels that tend to correlate with ionic strength. Therefore, the approach used to select maximum fluoride levels at high ionic strength is reasonable because it is based on relationships between fluoride concentration and ionic strength observed in material degradation rate sensitivity analyses at varying incoming water compositions. In addition, on the basis of its review of information provided in BSC Section 6.10.3[a] (2005ad), the NRC staff notes that DOE's maximum abstracted fluoride levels are set conservatively high when compared to fluoride levels observed in model output from the degradation rate sensitivity analyses at varying incoming water compositions. At low ionic strengths, the fluoride concentration either remains in the vicinity of the concentration of the incoming liquid or decreases due to mineral precipitation. Therefore, setting maximum fluoride concentrations to those observed in pore waters from the Topopah Spring welded tuff {i.e., 4.8 mg/L [0.00025 molal]} is reasonable at low ionic strengths because this is the maximum possible fluoride concentration that can occur in the waste package without water loss due to degradation reactions. On the basis of this evaluation, the approach DOE used to set maximum fluoride levels is reasonable and will not result in an underestimation of risk.

Summary of NRC Staff's Review of In-Package Chemical and Physical Environment

The NRC staff determines that, in modeling the in-package chemical and physical environment, DOE appropriately incorporated design features of commercial spent nuclear fuel (SNF) and codisposal waste packages. DOE used reasonable conceptual and mathematical models and assumptions to simulate geochemical interactions between fluids, gases, and internal components of the waste package and generate abstractions for pH, ionic strength, and fluoride concentration. DOE used technically defensible data to establish initial and boundary conditions for model simulations. These data included the thermodynamic properties of solids, gases, and aqueous species; incoming water chemistries; and the compositions, surface areas, and degradation rates of waste forms and material components of the waste package. The NRC staff notes that model simulations were applied over the full range of environmental conditions that might be expected inside breached waste packages. DOE appropriately approximated model uncertainty for propagation into the TSPA code by performing sensitivity analyses to assess the effects of uncertain thermal-hydrologic-chemical parameters on model outputs. Parameters with significant effects on model outputs (pCO$_2$ for pH and liquid influx rate and relative humidity for ionic strength) were incorporated as independent variables in the model abstractions. Parameters with smaller effects (material degradation rates) were used to quantify model uncertainty for propagation to the TSPA model. DOE provided support for the in-package chemistry pH abstractions by comparing predicted secondary mineral phase formation and estimated pH ranges to observations from natural soils and groundwater, natural analogs, and laboratory experiments. Support for the in-package chemistry ionic strength abstractions was provided by comparing predicted high ionic strengths to observations from natural groundwater and laboratory experiments. Therefore, on the basis of this review, DOE's abstraction and TSPA model implementation of the in-package chemical and physical environment are reasonable.

2.2.1.3.4.3.2 Waste Form Degradation

This section describes the NRC staff's review of DOE's abstraction and Total System Performance Assessment (TSPA) model implementation of radionuclide mobilization from waste form degradation. This radionuclide mobilization determines the quantity of radionuclides that may be transported by water from the solid waste form and eventually to the accessible environment. The waste form types include commercial spent nuclear fuel, high-level waste glass, and DOE spent nuclear fuel, as described in Safety Analysis Report (SAR) Section 1.5.1.

Commercial spent nuclear fuel is composed of irradiated fuels from pressurized water reactors and boiling water reactors. High-level waste glass is made by melting high-level radioactive materials with silica and/or other glass-forming chemicals and then solidifying them. DOE spent nuclear fuel (including naval spent nuclear fuel) comes from a range of high-level waste generators, from noncommercial reactors, and from the use of radioactive material that encompasses a variety of fuel types. On the basis of the significance to risk in the performance assessment calculations, the NRC staff's review focused on the inventory of radionuclides and radionuclide distribution in commercial spent nuclear fuel; degradation of commercial spent nuclear fuel; degradation of high-level waste glass; degradation of DOE spent nuclear fuel, naval spent nuclear fuel, and cladding; and associated model and data uncertainties, including waste form degradation under disruptive scenarios and microbial effects. Each waste form has its specific radionuclide inventory. In the nominal scenario, the waste form degrades as it dissolves after the cladding, if any, corrodes and fails in the aqueous environment. In the seismic or igneous scenarios, mechanically or thermally assisted degradation could also occur. For waste form degradation abstractions in the TSPA code, the input information includes the design description of the waste package, the waste form, the waste package internals, and in-package water chemistry and temperature. The output from this section includes waste form mobilization rates to assess engineered barrier system radionuclide transport.

SAR Sections 2.3.7.1–2.3.7.4, 2.3.7.6–2.3.7.9, 2.4, and associated references summarized DOE's model abstractions and related features, events, and processes (FEPs) related to the degradation of commercial spent nuclear fuel and cladding, high-level waste glass, and DOE spent nuclear fuel (including naval spent nuclear fuel).

Inventory of Radionuclides and Radionuclide Distribution in Commercial Spent Nuclear Fuel

The NRC staff's review of DOE's inventory data (SAR Section 1.5.1), in terms of weight, volume, and package design, for each waste form, is discussed in Technical Evaluation Report (TER) Sections 2.1.1.2.3.4.1, 2.1.1.2.3.4.2, and 2.1.1.2.3.5.1. More than 100 radionuclides may be collectively present in the waste package at the time of repository closure. Among them, a total of 32 isotopes of 18 elements were selected as important radionuclides to potential dose for scenario classes involving groundwater transport.

NRC Staff's Review

The NRC staff's evaluation of the radionuclide inventory incorporated DOE's design features of the waste forms in the waste package (SAR Section 1.5.1). The design features include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics. The NRC staff independently evaluated (i) the total mass of the waste form and (ii) the long half-lived radionuclides that should be considered in the repository as summarized in NRC/Center for Nuclear Waste Regulatory Analyses (CNWRA®) reports (Leslie, et al., 2007aa; Jain, et al., 2004aa; Manaktala, 1993aa). In these independent evaluations, the NRC staff reviewed open literature information. DOE appropriately selected the total mass inventory and the long-lived radionuclides because this information is supported by data and models and a significant underestimation of dose exposure will not occur.

The NRC staff evaluated DOE's radionuclide inventory calculations by comparing the results with other published results and with independent calculated inventory histories for times from 1 to 1 million years (Leslie, et al., 2007aa). The trends in the inventories were compared with those published elsewhere (e.g., Roxburgh, 1987aa). As expected, most radionuclide inventories decrease with increasing times, some remain relatively constant over long periods of time (those with long half-lives), and others increase with time (daughters in a decay chain). The NRC staff selected 43 radionuclides from hundreds of radionuclides present in commercial spent nuclear fuel using radionuclide screening processes. Screening criteria included half-lives, solubilities, and radiotoxicities of the radionuclides. Similar verifications were done for high-level waste glass and DOE (and naval) spent nuclear fuel.

Most radionuclides, and essentially all of the rare earth and actinide radionuclides (e.g., plutonium isotopes), are retained in the UO_2 matrix. Transition metals and fission products (e.g., technetium) are partly partitioned into metallic phases embedded in the matrix, spent nuclear fuel grain boundaries, and gap region (i.e., the interface between the pellets and the cladding). The NRC staff independently evaluated the distribution of radionuclides in the matrix and the accumulated radionuclides in the gap and grain boundaries. The NRC staff relied on open literature information and calculated radionuclide concentrations as a function of time, as earlier summarized in NRC/CNWRA reports (Leslie, et al., 2007aa; Jain, et al., 2004aa; Manaktala, 1993aa). The measured gap and grain boundaries in open literature include work by Bremier, et al. (2000aa), Gray (1999aa), Johnson and Tait (1997aa), and Lassmann, et al. (1995aa), and the changes in the radionuclide inventories were calculated using ORIGEN-ARP 2.00 (Bowman and Leal, 2000aa).

DOE reasonably represented the radionuclide distribution in the spent nuclear fuel matrix, because the inventory was based on experimental data and standard analytical techniques the industry uses to establish the distribution of actinides, transition metals, and fission products, and is consistent with values documented in the open literature.

Degradation of Commercial Spent Nuclear Fuel

DOE reported that if the waste package and cladding are breached, oxidation and dissolution of the commercial spent nuclear fuel matrix may occur. If the temperature exceeds approximately 100 °C [212 ºF], solid-state oxidation or hydration will occur, depending on the relative humidity. Commercial spent nuclear fuel dissolves by oxidative reaction of the UO_2 matrix in humid air or in solution at temperatures less than approximately 100 °C [212 ºF]. Oxidation and hydration occur faster than dissolution. Oxidation, hydration, and dissolution can be preferentially enhanced along grain boundaries. In DOE's commercial spent nuclear fuel degradation model (BSC, 2004ah), the high end of the dissolution rate range was obtained from tests in fast-flowing carbonate solutions. The low end of the dissolution rate range was obtained from commercial spent nuclear fuel rod segment tests under dripping groundwater conditions with precipitates deposited and failed cladding present. Un-irradiated UO_2 was also tested because there is no significant difference between the dissolution rates of un-irradiated UO_2 and commercial spent nuclear fuel under air-saturated groundwater conditions. Data from long-term immersion and dripping water tests up to 8.7 years in duration were included in the model development.

On the basis of these data analyses, DOE presented quantitative models and model parameter values for (i) the instantaneous release of radionuclides from the gap and grain boundaries and (ii) matrix dissolution inducing slow long-term radionuclide releases (SAR Sections 2.3.7.7, 2.4.2.2, and 2.4.2.3). Mean fractional matrix dissolution rates were 5×10^{-4} to 6×10^{-3} year^{-1} at pH of 5.5–8.0 and a temperature range of 25–90 °C [77–194 °F] under wet conditions, according to DOE Enclosure 5 (2009an). Fission products and activation products were released with the matrix dissolution. Actinide releases may be controlled by solubility limits of dissolved radionuclides and also may be affected by colloids. DOE addressed uncertainties of its models and parameter values.

In DOE's TSPA model, high-solubility fission products and activation products (e.g., I-129 and Tc-99) are released from the waste form at rates controlled by (i) the decay of radionuclide inventory of each waste package and (ii) waste package failure rate (e.g., SAR Section 2.4.2.2.3.2.1). The waste package failure rate is related in series to waste form dissolution rate and radionuclide diffusion rate. The slowest rate among the three controls the rate of release. DOE supported its model by stating that the dissolution rates of waste forms, including commercial spent nuclear fuel, are sufficiently faster (e.g., hundreds to a few thousand years) than the time intervals of each waste package failure. Low-solubility radionuclides (e.g., plutonium isotopes) are released at rates controlled mainly by the concentration limits of dissolved or colloidal species.

NRC Staff's Review of the Initial Condition of Spent Nuclear Fuel at Receipt

The NRC staff notes that the model abstraction of the degradation of commercial spent nuclear fuel reasonably incorporates DOE's waste form design features (which include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics; SAR Section 1.5.1), in that the commercial spent nuclear fuel conditions at receipt will not be altered during transportation and interim storage, as outlined in DOE Enclosure 3 (2009ax). DOE appropriately assumed that the pressure of the residual water vapor inside the transportation, aging, and disposal canister would not be sufficiently high to degrade the commercial spent nuclear fuel matrix by grain boundary hydration. The NRC staff's evaluation is based on DOE's use of standard vacuum drying procedures in packaging waste. The vacuum drying lowers the residual water vapor, resulting in no matrix degradation by grain boundary hydration. In addition, the NRC staff notes from the staff's literature review that potential matrix disintegration by high burnup in the range of ~60–65 MWD/kgU (e.g., Finch, et al., 1999aa; NRC, 2008aa) is not likely (Spino, et al., 2003aa). Also, in the TSPA model, release out of the waste package to the invert from spent nuclear fuel dissolution is insensitive to the initial condition. The release is mainly controlled by either (i) the radionuclide inventory of each waste package and waste package failure rate or (ii) concentration limits. Therefore, DOE presented reasonable technical arguments that the pressure of residual water vapor inside the transportation, aging, and disposal canister would not be sufficiently high to disintegrate the commercial spent nuclear fuel matrix by grain boundary hydration, and the residual water vapor will not cause the release of radionuclides out of the waste package to the invert from spent nuclear fuel dissolution to be underestimated.

NRC Staff's Review of Releases From the Matrix and From the Gap and Grain Boundaries

The NRC staff evaluated DOE's processes and modeling for matrix dissolution and radionuclide release from the gap and grain boundaries. The NRC staff reviewed open literature information for this evaluation, including results from the NRC staff's independent modeling (NRC, 2008aa; Leslie, et al., 2007aa; Jain, et al., 2004aa). The NRC staff notes that DOE reasonably concluded that the release of high-solubility radionuclides (e.g., I-129 and Tc-99) will be at the same rate of oxidative UO_2 matrix dissolution, whereas release of low-solubility radionuclides (e.g., plutonium isotopes) may be limited by solubility. The DOE conclusions are consistent with laboratory test results (Wilson and Gray, 1990aa).

The literature (Shoesmith, 2000aa) suggests that the UO_2 matrix would dissolve oxidatively in the oxidizing environment expected at the proposed Yucca Mountain repository. This is consistent with the alteration process of natural analog uraninite (BSC, 2004ah). DOE's mathematical models are empirical. DOE identified important environmental and commercial spent nuclear fuel parameters controlling the dissolution rate. Those parameters included oxygen partial pressure, carbonate concentration, temperature, pH, and the surface area of the matrix contacted by water. These parameter values were obtained from accelerated test results in oxidizing environments, with consideration of data uncertainties. These parameters and their values are reasonable, because they were derived from tests that are based on standard practice. In addition, the NRC staff performed independent evaluations (Leslie, et al., 2007aa) that are consistent with DOE's results. DOE reasonably assumed that radionuclide release from the gap and grain boundaries is rapid because these radionuclides are not atomically bound in the matrix. Therefore, the NRC staff notes that DOE conservatively presented the conceptual and mathematical models for radionuclide releases that are faster than those expected to occur from matrix degradation and from the gap and grain boundaries.

NRC Staff's Review of Environmental Conditions Inside a Waste Package

On the basis of DOE's in-package chemistry models (BSC, 2005ad) and the NRC staff's independent analyses (Leslie, et al., 2007aa; NRC, 2008aa, 1996ab), the NRC staff notes that DOE accounted for the range of environmental conditions expected inside the breached waste packages. DOE based the commercial spent nuclear fuel degradation model on pure carbonate solutions in the concentration range of 2×10^{-4} to 2×10^{-2} mol/L. This is conservative because carbonate solutions lead to faster matrix dissolution in a waste package (NRC, 2008aa). The NRC staff's independent review and analysis of open literature information (Leslie, et al., 2007aa; NRC, 2008aa, 1996ab) suggest that dissolved constituents, such as calcium and silica, may reduce the dissolution rate. Slow dripping groundwater, partial protection by failed cladding, and iron corrosion products may also reduce the dissolution rate. As explained in DOE Enclosure 16 (2009ax), fast spent nuclear fuel dissolution in pure carbonate solutions may cause more actinide release associated with colloids. The NRC staff reviewed DOE's derived dissolution processes of spent nuclear fuel in terms of the rates of dissolution and colloid formation of actinides, such as plutonium. On the basis of DOE's conservative approach using carbonate solutions, the NRC staff's independent review of dissolution data in open literature (NRC, 2008aa, 1996ab), and the TSPA's dose consequence, the magnitude of this increased actinide release is not significant, because realistic matrix dissolution rates are lower. In the Total System Performance Assessment (TSPA) model, however, for high-solubility radionuclides, such as technetium, the release rate is controlled by the radionuclide inventory of each waste package and waste package failure rate. In this rate-controlled TSPA model, the slow realistic dissolution rates may result in increased release rates because multiple waste package failures at low dissolution rates may have a greater contribution to the release. The

NRC staff's confirmatory analyses of the effects of multiple waste package failures suggest that the magnitude of the increase in release is minimal and insignificant. The NRC staff compared the release at different dissolution rates per unit area assuming spherical spent nuclear fuel particles from single- or multiple-remnant-failed waste packages at slow realistic dissolution rates. On the basis of this comparison, the NRC staff notes that, within the range of a factor of 10 decrease in dissolution rates (NRC, 2008aa), the addition of release by slow dissolution from remnant failed waste packages is insignificant.

NRC Staff's Review of Alternative Models for Matrix Dissolution

In BSC Section 6.4.2 (2004ah), DOE presented an electrochemical model and a surface complexation model as alternative models for the dissolution of the commercial spent nuclear fuel. The electrochemical model describes the process of the matrix dissolution by electric current flow under oxidizing conditions, and the surface complexation model describes the dissolution process by carbonate complexation. On the basis of its review, the NRC staff notes that DOE's model results for dissolution rate are consistent with the basecase results within the uncertainty limits. In addition, the NRC staff performed independent assessments of the processes involved in these two alternative models by reviewing open literature information (NRC, 2008aa, 1996ab). On the basis of its literature review, the NRC staff notes that DOE's basis and justification for the alternative models are reasonable. DOE's results are consistent with the NRC staff's independent evaluations (NRC, 2008aa, 1996ab).

Degradation of High-Level Waste Glass

DOE conceptually modeled high-level waste glass as being congruently dissolved for glass constituent elements and radionuclides at relative humidity greater than or equal to 44 percent. At lower relative humidity, the glass dissolution rate is set to zero. Dissolution kinetics were considered to be chemically controlled by dissolved orthosilicic acid (H_4SiO_4). As glass reacts with solution and reaches saturation with respect to mineral phases, precipitation occurs on the glass surface. DOE's model was supported by dissolution studies conducted with a wide range of borosilicate glass compositions under various environmental conditions. On the basis of the data available from DOE's tests, DOE presented a quantitative model and model parameter values for the high-level waste glass dissolution process.

NRC Staff's Review

The model abstraction of the degradation of high-level waste glass incorporates DOE's waste form design features, which include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics (SAR Section 1.5.1). The NRC staff evaluated the processes and modeling that DOE presented for the dissolution of high-level waste glass. The NRC staff's evaluation was based on open literature information originally compiled by Leslie, et al. (2007aa).

The NRC staff notes that DOE reasonably assumed that the release of high-solubility radionuclides (e.g., I-129 and Tc-99) will be at the same rate of matrix dissolution, especially at repository relevant pH 5–8 in diluted in-package water chemistry at lower temperatures of 25-90 °C [77–194 °F] after waste package failure (BSC, 2005ad; Leslie, et al., 2007aa). The NRC staff notes that the basis for this assumption is that higher solubility radionuclides will not tend to form immobile precipitates. As addressed in Technical Evaluation Report (TER) Section 2.2.1.3.4.3.1, the NRC staff notes that DOE appropriately assessed this environmental condition (BSC, 2005ad). DOE considered both sodium- and calcium-based pore waters and data on the matrix dissolution under immersion, dripping groundwater, and vapor conditions. The data from these conditions were consistent with each other, and some accelerated tests such as fast-flowing water tests were also used to determine model parameter values. The NRC staff notes that the condition of mild aqueous chemistry used in the assessment would persist for the dissolution because the waste package would fail after several ten (for stress corrosion cracking) to hundred (for general corrosion) thousand years. DOE's assertion that release of low-solubility radionuclides (e.g., plutonium isotopes) may be limited by solubility is reasonable. These two release modes and models are consistent with laboratory data presented by DOE and in the literature for borosilicate high-level waste glass in simulated Yucca Mountain in-package water, as described in BSC (2005ad), under immersion, dripping groundwater, and vapor conditions.

DOE reasonably included in mathematical models important environmental and high-level waste glass parameters controlling the dissolution rate. The mathematical models are reasonable because they are consistent with an independent assessment of the dissolution rate based on open literature data (Leslie, et al., 2007aa). DOE quantitatively modeled the release rate of radionuclides as a function of surface area of glass contacted by water, intrinsic glass dissolution rate (i.e., release rate of boron as an indicator), pH, activation energy for temperature dependence, and the extent of orthosilicic acid saturation in solution with the glass. The dissolution rates in acidic and alkaline regimes of pH were separately modeled. At 100–250 °C [212–482 °F], a fixed pH was used. The mean fractional dissolution rates are 2×10^{-5} to 4×10^{-3} year^{-1} at pH of 5.5–8.0 and temperature of 25–90 °C [77–194 °F] under wet conditions, according to DOE Enclosure 5 (2009an).

DOE obtained model parameters and data uncertainties from suitable data, as documented in SAR Section 2.3.7.9. The equation used by DOE to calculate the area of glass surface contacted by water as glass dissolves accounts for (i) an increase in surface area from thermal and mechanical cracking, water access and reactivity with water and (ii) a loss in the surface area due to dissolution, as detailed in DOE Enclosure 2 (2009ax) and DOE Enclosure 2 (2009cz). The increased surface area leads to increased release of radionuclides out of the waste package to the invert from high-level waste glass dissolution. In DOE's model this increasing factor of surface area is expressed as "exposure factor." The NRC staff notes that model or data uncertainties for other processes were appropriately discussed with respect to the distributions of model parameters (e.g., the extent of orthosilicic acid saturation) and conservatisms. DOE's uncertainty assessment is reasonable, because the uncertainties of parameter values did not significantly affect the radionuclide release out of the waste package. Model support was also discussed with respect to long-term field tests and natural analog basalt glass. The NRC staff considers basalt glass to be an appropriate analog to high-level waste glass because of similarities in composition, reaction conditions, and transformation of the glass matrix during alteration into a range of minerals. Jantzen, et al. (2008aa) reviewed and analyzed existing field tests of high-level waste glass buried for ~24 years; this review further supports the SAR model and data for high-level waste glass dissolution. Jantzen, et al. (2008aa) showed superior or equivalent performance of this burial glass in unsaturated and saturated sediments, compared to saturated accelerated laboratory tests that are the basis for DOE's model.

Literature data and discussions on cracking (or pitting) of various glasses in more aggressive solutions (Pulvirenti, et al., 2006aa; Morgenstein, et al., 1999aa) did not show significant increase in dissolution rates. The test solutions used by Pulvirenti, et al. (2006aa) and Morgenstein, et al. (1999aa) were aggressive, unlike those expected in the repository, and many glasses considered were not based on borosilicates intended for disposal in the proposed repository. DOE also assessed the volume occupied by porosity in the altered layer during the glass hydration. The porosity may increase the glass volume in a confined canister, which may create stress that further fractures the glass. The calculated porosity volume was close to the elemental mass percentage of soluble elements in the glass. Therefore, DOE considered that isovolumetric hydration would occur. For high-solubility radionuclides in TSPA, the release rate is controlled by the radionuclide inventory of each waste package and waste package failure rate because the dissolution rate is faster, as discussed for the degradation of commercial spent nuclear fuel. The NRC staff determines that the increased dissolution (even with further cracking by any means) associated with data or model uncertainties would not be rate controlling in the release. The waste form dissolution rate is related in series to the waste package failure rate and the radionuclide diffusion rate. The slowest among the three rates, the waste package failure rate in this case, controls the release rate.

Therefore, DOE's conceptual and mathematical models for radionuclide releases from high-level waste glass degradation are reasonable. The NRC staff also notes that DOE reasonably accounted for the range of environmental conditions expected inside breached waste packages, because DOE's in-package water chemistry models (BSC, 2005ad) were developed using current analytic techniques and thermodynamic approaches accepted in the technical community. In addition, the NRC staff's independent models (Leslie, et al., 2007aa) predicted similar conditions considering immersion, dripping groundwater, and vapor environments.

As an alternative model for high-level waste glass dissolution, DOE presented a dissolution model without considering the extent of orthosilicic acid saturation (i.e., affinity) (BSC, 2004ai). On the basis of its review of open literature information (e.g., Leslie, et al., 2007aa; BSC, 2004ai), the NRC staff notes that the alternative model is conservative and consistent with available data/analyses and current scientific understanding.

Degradation of DOE and Naval Spent Nuclear Fuel, and Cladding

DOE divided its spent nuclear fuel into 34 distinct forms. Except for naval spent nuclear fuel, these DOE spent nuclear fuel types were modeled as degrading instantaneously upon waste package breach. Commercial spent nuclear fuel waste packages were used to represent the naval spent nuclear fuel waste packages for all scenario classes, because radionuclide release rates from the naval spent nuclear fuel waste packages were predicted to be considerably lower than from commercial spent nuclear fuel waste packages (BSC, 2004ao). Data uncertainties were discussed with respect to the conservatisms used.

NRC Staff's Review

DOE reasonably assumed that DOE spent nuclear fuel will degrade instantaneously, because this assumption would not underestimate the radiological consequences. The NRC staff notes that DOE's approach to model the naval spent nuclear fuel as commercial spent nuclear fuel is reasonable. This modeling assumption would not underestimate the radiological consequences, because the naval spent nuclear fuel is more robust and would release less radionuclides (BSC, 2004ao). This is also supported by the TSPA model. For high-solubility radionuclides, the release rate is controlled mainly by the inventory of each waste package and waste package failure rate; for low-solubility radionuclides, the release rate is controlled mainly by the concentration limits of dissolved species or colloids. The TSPA abstraction provides results consistent with output from detailed process-level models and/or empirical observation of DOE spent nuclear fuel characteristics (DOE, 2003ad).

The assumption of instantaneous degradation for DOE spent nuclear fuel may result in greater and faster colloidal release in a shorter period, a possibility addressed in DOE Enclosure 1 (2009ax) and DOE Enclosure 1 (2009cz). This is similar to the case of instantaneous degradation of the waste form under igneous intrusive scenarios as described in the following subsection (Other Model and Data Uncertainties – Waste Form Degradation under Disruptive Scenarios). More realistically, after the instantaneous degradation, the waste form would be altered into other solid forms, such as further oxidized UO_2, other oxide metal compounds, or hydrolysis or precipitation products (SNL, 2008ak, 2007ag), before being slowly dissolved, as described in DOE Enclosure 1 (2009cz). DOE considered colloid formation during the instantaneous degradation and subsequent alteration of the waste form. The NRC staff notes that model support for the quantitative information on colloid concentrations during these processes is reasonable. As outlined in DOE Enclosure 1 (2009ax), DOE Enclosure 1 (2009cz), and DOE Enclosure 1 (2009db), DOE showed that faster dissolution does not necessarily result in greater plutonium colloid concentration in the bulk solution, because the majority of the plutonium remains in a residue (altered) layer at the reaction front.

DOE assumed that the zircaloy and stainless steel commercial spent nuclear fuel cladding failed upon emplacement. Therefore, degradation of commercial spent nuclear fuel cladding was not included in the TSPA analysis. The effect of the naval spent nuclear fuel structure on the release and transport of radionuclides was treated separately from other DOE spent nuclear fuel types in the assessment. DOE spent nuclear fuel was conservatively assumed to degrade instantaneously. The naval spent nuclear fuel degrades more slowly than the commercial spent nuclear fuel, and therefore the naval spent nuclear fuel can be represented by commercial spent nuclear fuel waste packages in TSPA analysis. DOE documented the technical basis for cladding behavior in the repository briefly because DOE conservatively assumed that the zircaloy and stainless steel commercial spent nuclear fuel cladding failed at time of emplacement. This assumption is conservative because it does not underestimate the barrier performance of the fuel cladding.

Other Model and Data Uncertainties

Waste Form Degradation Under Disruptive Scenarios

DOE stated that under the seismic scenario class, stress corrosion cracking of the waste package would occur earlier than waste package failure in the nominal case. Seismic response (motion and rockfall) could damage the drip shield and waste package, resulting in earlier stress corrosion cracking. The waste form dissolves in the diffused-in water vapor through the stress corrosion cracks, as in the nominal case. The igneous scenario class includes eruptive and intrusive events. In the volcanic eruption case, the impacted waste form was transported to the surface; this case does not involve groundwater transport and is not discussed in this Technical Evaluation Report (TER) chapter. In the igneous intrusive case, the waste form was assumed to be rapidly altered at expected elevated temperatures and made available to groundwater.

NRC Staff's Review

The NRC staff reviewed the degradation models implemented in the Total System Performance Assessment (TSPA) code abstraction to confirm that they provide consistent results with the output from the detailed process-level models and/or empirical observations on the characteristics of commercial spent nuclear fuel and high-level-waste glass, as described in this TER section. DOE presented the bounding assumption regarding radionuclide release from all waste forms under igneous intrusive conditions in Safety Analysis Report (SAR)

Section 2.3.11.3.2.4. DOE assumed that all waste forms instantaneously degrade to be mobilized for release. These instantaneous degradation models are conservative and bounding in terms of soluble radionuclide release. As discussed earlier, in the TSPA model the release from the failed waste package is insensitive to the fast degradation rate of waste form. The release rate is controlled mainly by the inventory of each waste package and waste package failure rate for high-solubility radionuclides, and concentration limits of dissolved species or colloids for low-solubility radionuclides. The assumption of instantaneous degradation under igneous intrusive conditions may result in more and faster colloidal release in a shorter period, as addressed in DOE Enclosure 1 (2009ax) and DOE Enclosure 1 (2009cz). For the nominal scenario case, DOE presented concentrations of 1.2×10^{-8} to 6.2×10^{-4} g/L [1.2×10^{-5} to 6.2×10^{-1} ppm (part per million)] for irreversible colloids from commercial spent nuclear fuel, as detailed in DOE Enclosure 16 (2009ax), and 2.7×10^{-6} to 1.4×10^{-5} g/L [2.7×10^{-3} to 1.4×10^{-2} ppm] for high-level-waste glass colloids (SAR Section 2.3.7.11.3). DOE's instantaneous degradation assumption is conservative because the waste form can be altered into other solid forms, such as oxidized UO_2, other oxide metal compounds, or hydrolysis or precipitation products (SNL, 2008ak, 2007ag), before being slowly dissolved, as detailed in DOE Enclosure 1 (2009cz).

DOE considered colloid formation during the instantaneous degradation and subsequent alteration of the waste form. The model support for the quantitative information on colloid concentrations during these processes is reasonable. In DOE Enclosure 1 (2009ax), DOE Enclosure 1 (2009cz), and DOE Enclosure 1 (2009db), DOE showed that faster dissolution does not necessarily result in greater plutonium colloid concentration in the bulk solution, because the majority of the plutonium remains in a residue (altered) layer at the reaction front.

Microbial Effects

DOE did not specifically address the microbial effects potentially affecting the dissolution of spent nuclear fuel and high-level waste glass. However, DOE's models indirectly incorporate the effects to the extent that the models are consistent with natural analog and field test results. The NRC staff notes that the analog data or field test results could have been affected by the potential presence of microbial effects, compared with the laboratory test results.

NRC Staff's Review

The NRC staff reviewed the waste form degradation that was partially tested in solutions under projected repository conditions, along with field tests and analog studies. These aqueous environments, especially field test or analog environments, may have contained organic byproducts or microbes. DOE's models also include the characteristics of natural analogs of the waste form or field test results. No indication of microbe effects (e.g., lowering pH) were reported from these literature data (BSC, 2004ah, 2004ai). Therefore, DOE's approach to the effects of organic byproducts or microbes on the waste form degradation is reasonable. Screening arguments are also found in excluded FEP 2.1.02.10.0A, Organic/Cellulosic Materials in Waste (SNL, 2008ab). Excluded features, events, and processes are not explicitly discussed in this chapter. The NRC staff notes that the technical basis for DOE's exclusion of features, events, and processes is reasonable in TER Section 2.2.1.2.1.3.2.

Integration in the Engineered Barrier System Radionuclide Transport Abstraction

The waste form mobilization abstraction provides radionuclide inventory, mobilized radionuclides, and waste form colloids to the engineered barrier system radionuclide transport model. The NRC staff notes that DOE has appropriately deployed the waste form mobilization using GoldSim in the waste form domain. The NRC staff confirmed that the GoldSim results are consistent with DOE's description and quantitative assessments of waste form lifetimes in the waste form domain.

Summary of NRC Staff's Review of Waste Form Degradation

DOE appropriately incorporated design features of the waste form. The design features include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics. The NRC staff reviewed DOE's proposed dissolution processes of waste forms and notes that the following DOE conclusions are reasonable: (i) the dissolution process of commercial spent nuclear fuel is based on oxidative dissolution of the UO_2 matrix and (ii) the dissolution process of high-level-waste glass is based on orthosilicic acid release. The NRC staff also notes that the models for DOE spent nuclear fuel dissolution are bounded by the assumption of instantaneous dissolution and that modeling naval spent nuclear fuel as commercial spent nuclear fuel is conservative. In the models, DOE reasonably includes important environmental and waste form parameters controlling the dissolution rate (e.g., pH, temperature, solution chemistry, cracking, and hydration). DOE's models are reasonable and were supported by laboratory and/or field test results. DOE obtained sufficient data to vary environmental and waste form parameters. The model uncertainties were assessed with alternative models; the data uncertainties were assessed with sufficient data obtained under various environmental and waste form conditions. The TSPA model considers uncertainties associated with (i) conservative, fast dissolution of commercial spent nuclear fuel under nominal conditions and (ii) the instantaneous dissolution of all waste forms under igneous conditions and DOE spent nuclear fuel under nominal conditions. DOE appropriately implemented its TSPA code with the model abstraction of radionuclide mobilization from the waste form.

2.2.1.3.4.3.3 Concentration Limits

This section discusses the NRC staff's review of DOE's abstraction and Total System Performance Assessment (TSPA) model implementation of dissolved radioactive element (radioelement) concentration limits, as described in Safety Analysis Report (SAR) Section 2.3.7.10. For significant periods of time in the performance assessment model, concentration limits exert strong controls on the concentrations in water of dose-important radioelements—particularly neptunium and plutonium—and, thus, on the release rates of those elements' isotopes from the engineered barrier system. These limits are based on chemical equilibrium relationships between the dissolved element and solid substances containing the element. The abstraction calculates, on the basis of water chemistry, maximum concentrations that limit how much of the total mass of a radioelement may remain dissolved in solution in the waste form domain, the corrosion products domain, and the invert (the drift floor, consisting of crushed rock). In the waste form domain, the radionuclide concentration calculated from the waste form degradation abstraction is reduced if it, along with the concentrations of other isotopes of the same element, exceeds the calculated concentration limit. The concentration limit comparison is also implemented for the radionuclide transport abstraction in the corrosion products domain and the invert. In each case, the remaining radionuclide mass is retained in the domain as a precipitated mass that is available for re-dissolution whenever the concentration is below the concentration limit. The inputs to the concentration limits abstraction are geochemical characteristics in the domain of the water from the in-package chemistry abstraction (waste form domain; TER Section 2.2.1.3.4.3.1) or the engineered barrier system radionuclide transport abstraction (corrosion products domain and invert; TER Section 2.2.1.3.4.3.5), and gas from the engineered barrier system chemical environment abstraction (TER Section 2.2.1.3.3). The outputs from the TSPA model abstraction are the concentration limits used in the engineered barrier system radionuclide transport abstraction (TER Section 2.2.1.3.4.3.5). The actual application of the concentration limits and retention of the precipitated mass is calculated in the GoldSim computer code, as outlined in GoldSim Technology Group, pp. 253–255 (2006aa). Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and Engineered Barrier System is an included FEP encompassing the abstraction evaluated in this section. The related excluded features, events, and processes are addressed in TER Section 2.2.1.2.1.3.2.

To evaluate DOE's abstractions of radioelement concentration limits, the NRC staff reviewed SAR Section 2.3.7, the analysis model report on concentration limits (SNL, 2007ah), and DOE's responses to the NRC staff's requests for additional information (DOE, 2010aj; 2009ax,ay,cz,da,db,dc). The NRC staff also relied on the technical literature on solubility limits and the application of solubility limits in performance assessments, the NRC staff's independent solubility limit evaluations (e.g., Murphy and Codell, 1999aa; Mohanty, et al., 2003aa), and the NRC staff's independent laboratory studies (e.g., Prikryl, 2008aa).

Overall Abstraction Approach

DOE's abstraction for concentration limits calculates concentration limits for plutonium, neptunium, uranium, thorium, americium, tin, and protactinium using lookup tables that define values (in mg/L, a unit that is approximately equivalent to parts per million, or ppm) as functions of pH and fCO_2 (i.e., CO_2 fugacity). For radium, the value is specified as a constant that depends on the range in which the pH value falls. For technetium, carbon, iodine, cesium, strontium, selenium, and chlorine, no concentration limit is applied; this abstraction, therefore, does not affect their release rates from the engineered barrier system.

For plutonium, neptunium, uranium, thorium, americium, tin, and protactinium, DOE determined the concentration limit value from the lookup table for each timestep in a realization. Uncertainty is incorporated into the abstraction by sampling two uncertainty terms that are then added to (or subtracted from) the value derived from the lookup table. No further uncertainty is applied to the determined radium value. Concentration limits are treated the same for nominal and disruptive events, with the exception of the uranium abstraction in the igneous intrusive case (see uranium discussion in the Concentration Limits Parameters section).

NRC Staff's Review

DOE's overall approach to impose concentration limits is appropriate because it is consistent with standard thermodynamic geochemical principles and uses consistent and reasonable assumptions. The use of pure phase solubilities to constrain radioelement concentrations at the source is an accepted approach in performance assessments for radioactive waste disposal (e.g., Nuclear Energy Agency, 1997aa; Leslie, et al., 2007aa). When faced with uncertainty regarding the solid phase with which to model a solubility limit, DOE reasonably chose the solid phase that would result in higher dissolved concentrations (e.g., hydrated PuO_2 instead of anhydrous PuO_2).

Chemical Environment for Concentration Limits

For the waste form domain, DOE's in-package chemistry abstraction provides pH, ionic strength, and fluoride concentration to the concentration limits abstraction at each timestep, and CO_2 fugacity (fCO_2) is obtained from the engineered barrier system chemical environment abstraction. The same ionic strength and fCO_2 are used in the corrosion products domain, but pH is calculated from a formula that involves fCO_2 and the aqueous uranium concentration; this pH abstraction is based on the competitive surface complexation model discussed in Technical Evaluation Report (TER) Section 2.2.1.3.4.3.5. In the corrosion products domain, according to DOE Enclosure 3 (2009ay) and DOE Enclosure 2 (2009da), the abundant mass of products of stainless steel corrosion controls pH to a relatively narrow range of 7.0 to 8.4. For the invert, when there is no flow from the waste package, pH, ionic strength, and fCO_2 are obtained from the engineered barrier system chemical environment abstraction; when there is advective flow out of the waste package, according to DOE Enclosure 7 (2009ax), pH and ionic strength in the invert are the same as in the corrosion products domain.

NRC Staff's Review

The NRC staff notes that DOE used appropriate tools to model concentration limits, including the important geochemical parameter inputs (e.g., pH and fCO_2) affecting the solubility model outputs. Concentration limits in the waste form domain are functions of chemical parameters developed by the in-package chemistry model (TER Section 2.2.1.3.4.3.1). Chemical conditions for concentration limits in the corrosion products domain were reasonably modeled using the surface complexation model, and the results of that model were supported by comparison with DOE's independent modeling efforts in DOE Enclosure 2 (2009da).

An abundant secondary phase, such as steel corrosion products, can have an important influence on pH buffering in an environment with such high solid-to-water ratios. The NRC staff questioned DOE concerning the choices of stainless steel corrosion rates and their effects on in-package pH in the performance assessment. In DOE Enclosure 2 (2009db), DOE provided information that (i) supported its selection of the uncertainty range for the stainless corrosion rate on the basis of laboratory data and (ii) showed that plutonium isotope release rates, which are sensitive to the pH-dependent plutonium solubility limit, are insensitive to the stainless steel corrosion rate. The NRC staff verified DOE's analysis by reviewing corrosion rate data in the literature (Beavers and Durr, 1990aa; BSC, 2004ae; Glass, et al., 1984aa; McCright, et al., 1987aa). In addition, DOE showed in DOE Enclosure 3 (2009db) that conservatisms in (i) the treatment of the timing of radionuclide release after waste package breach, (ii) assumptions regarding flow within the waste package, and (iii) the lower pH range meant that DOE was unlikely to overestimate the effectiveness of stainless steel corrosion products in controlling pH. The NRC staff notes that this is reasonable, in that the abstraction would not result in underestimation of radionuclide release rates, and is consistent with the NRC staff's understanding of the abstraction.

On the basis of this review, DOE's TSPA integration of the concentration limit abstraction and its accounting for the range of geochemical environments expected in the waste package and invert are reasonable.

Concentration Limits Parameters

DOE calculated concentration limits for plutonium, neptunium, uranium, thorium, americium, tin, protactinium, and radium (using barium as an analog) assuming pure-phase solubility at equilibrium with solution using the geochemical modeling code EQ3NR. The solubility models were conducted at a range of pH, fCO_2 (CO_2 fugacity), and fluoride values. For plutonium, neptunium, uranium, thorium, americium, tin, and protactinium, the pH and fCO_2 dependencies were incorporated into the lookup tables, and the fluoride sensitivity was applied through an uncertainty term. SAR Section 2.3.7.10 described the technical bases for the limiting minerals selected for the solubility models. For most of the modeled elements, equilibrium with atmospheric oxygen was assumed; this assumption was modified for plutonium in the waste package. As mentioned previously, DOE did not apply a concentration limit for technetium, carbon, iodine, cesium, strontium, selenium, and chlorine.

NRC Staff's Review

DOE's use of no concentration limits for technetium, carbon, iodine, cesium, strontium, selenium, and chlorine is consistent with the lack of a strong technical basis for concentration limits and is conservative because it does not underestimate the radiological consequences of release of isotopes of these seven elements. Application of a concentration limit can only reduce the dissolved concentration of an element; therefore, ignoring this process can only increase radionuclide release rates or leave them unchanged.

The NRC staff notes that DOE used, in its chemical speciation and solubility models, appropriate thermodynamic data that had been evaluated for this purpose and were based on extensive, scholarly, international reviews. More detailed concentration limit evaluations for each element are in the following paragraphs.

Plutonium

On the basis of DOE's TSPA code dose modeling results, plutonium is a risk-significant radioelement. DOE's plutonium concentration limits abstraction is based on equilibrium geochemical modeling, but differs from other abstractions in that equilibrium with atmospheric oxygen was not assumed. Atmospheric oxygen would impose higher redox potentials that tend to lead to higher calculated dissolved plutonium concentrations. The NRC staff therefore focused on DOE's adjusted-Eh model, which assumes lower Eh than would be imposed by atmospheric oxygen. (Eh is a measure of a solution's oxidation potential, which may be described as the tendency of the solution to convert dissolved elements to higher oxidation states.) The adopted Eh-pH relationship for plutonium solubility models is more oxidizing than the line bounding a compilation of data from waters in contact with atmosphere, as outlined in SNL (2007ah), DOE Enclosure 9 (2009ax), and DOE Enclosure 7 (2009cz). For the typical pH and CO_2 conditions in the waste package (SAR Figures 2.3.7-19 to 2.3.7-21), the sampled plutonium solubility limit in the TSPA model ranges approximately from 0.006 to 0.5 mg/L (SAR Figure 2.3.7-29); uncertainty terms extend this range as much as ± two orders of magnitude (e.g., SAR Figure 2.3.7-38).

NRC Staff's Review

On the basis of information DOE provided and on an independent NRC staff's review of the scientific literature (e.g., Rai, et al., 1999aa; Neck, et al., 2007aa), the plutonium solubility abstraction was reasonably developed using Eh-pH conditions that bound the in-package environment. As illustrated in SNL (2007ah), DOE Enclosures 8 and 9 (2009ax), DOE Enclosures 5 and 6 (2009cz), and DOE (2010aj), nearly all compared plutonium experimental data lie within two standard deviations of the uncertainty in the pH-dependent solubility relationship. DOE appropriately supported the plutonium model results by showing that most calculated concentrations will be higher than the results from spent nuclear fuel dissolution tests.

Neptunium

DOE identified neptunium as an important risk contributor. In general, DOE modeled neptunium solubility limits assuming equilibrium with atmospheric oxygen, but used two different solid phases depending on conditions. The controlling solid phase for the invert is the oxidized phase Np_2O_5. The choice of the neptunium-controlling mineral in the waste package (waste form and corrosion products domains)—NpO_2 or Np_2O_5—depends on the corrosion status of the steel components. DOE stated that local reducing conditions during steel corrosion would promote precipitation of reduced NpO_2 over oxidized Np_2O_5. DOE noted that the literature suggests that, in the presence of reducing materials, NpO_2 is an appropriate solubility-limiting solid phase under most modeled conditions. DOE chose Np_2O_5 to limit neptunium concentration in the absence of reductants. A sodium neptunium carbonate is modeled at the high pH margin of the water chemistry range. For the typical pH and CO_2 conditions in the waste package (SAR Figures 2.3.7-19 to 2.3.7-21), the sampled neptunium solubility limit in the Total System Performance Assessment (TSPA) analysis ranges approximately from 0.02 to 11 mg/L for NpO_2 and from 0.3 to 180 mg/L for Np_2O_5 (SAR Figure 2.3.7-30); uncertainty terms extend this range to more than an order of magnitude above and below those ranges (e.g., SAR Figure 2.3.7-39).

NRC Staff's Review

DOE reasonably chose NpO_2 and Np_2O_5 as solubility-limiting solids, depending on oxidation-reduction conditions. In support of its neptunium model, DOE showed that the range of calculated neptunium concentrations, as a function of pH, exceeds the majority of concentrations from spent nuclear fuel dissolution tests. Furthermore, the mean-value curves for concentration limit versus pH are higher than all the spent nuclear fuel dissolution test concentration values (SAR Figure 2.3.7-39). In DOE Enclosure 12 (2009ax), DOE compared the model to project and literature data that, in general, suggested higher neptunium solubility limits but are within the 2σ model uncertainty incorporated in its analysis. DOE showed that the experiments that yielded especially high neptunium concentrations measured solubilities of metastable hydrous neptunium oxides, rather than the lower solubility anhydrous neptunium oxides expected on longer repository time scales. Therefore, the highest measured neptunium concentrations could be excluded from development of neptunium concentration limits for performance assessment. In addition, DOE considered an alternative conceptual model involving neptunium incorporation in secondary uranyl minerals resulting from spent nuclear fuel dissolution. DOE excluded the alternative conceptual model because of inadequate technical bases for its inclusion in performance assessment, which is consistent with a similar analysis the NRC staff previously made (Pickett, 2005aa). Consequently, the NRC staff notes that DOE's neptunium model is reasonable.

Uranium

DOE produced two different lookup tables for uranium, depending on the particular chemical environment. In most cases for commercial spent nuclear fuel packages, the hydrated uranyl oxide schoepite is the solubility-limiting solid. For the typical pH and CO_2 conditions in a commercial spent nuclear fuel waste package (SAR Figure 2.3.7-19), the sampled uranium solubility limit in the TSPA code ranges approximately from 1 to 100 mg/L [ppm], as described in SNL Figure 6.7-1 (2007ah). The associated uncertainty terms extend this range as much as ± an order of magnitude; for example, as detailed in SNL Figure 7-6 (2007ah). For codisposal packages in all scenarios and all packages in the igneous intrusion scenario, and for the invert in all scenarios, the uranyl silicate sodium-boltwoodite and $Na_4UO_2(CO_3)_3$ are included in the solubility models. For the typical pH and CO_2 conditions in a codisposal waste package (SAR

Figures 2.3.7-20 and 2.3.7-21), the sampled uranium solubility limit in the TSPA code ranges approximately from 1 to more than 10,000 mg/L (SAR Figure 2.3.7-31). The associated uncertainty terms extend this range as much as ± an order of magnitude; for example, as illustrated in SNL Figure 7-6 (2007ah).

NRC Staff's Review

The NRC staff notes that DOE chose reasonable solubility-limiting solid phases for the uranium solubility limit model. Secondary uranyl minerals such as uranophane (Prikryl, 2008aa) were conservatively excluded in favor of schoepite and sodium-boltwoodite. These excluded minerals could control uranium to even lower concentrations under certain chemical conditions. Laboratory and natural analog studies DOE cited (SNL, 2007ah) support these choices. A relatively soluble sodium uranyl carbonate was modeled at highest pH and fCO_2. DOE appropriately compared model results with uranium concentrations measured in spent nuclear fuel dissolution tests; this comparison shows that most measured uranium concentrations plot near or below the model solubility limit curves, as shown in SNL Figure 7-6 (2007ah). Therefore, DOE's uranium model is reasonable.

Thorium, Americium, Protactinium, Radium, and Tin

In developing lookup tables for the other modeled actinides—thorium, americium, and protactinium—DOE chose solubility-limiting phases on the basis of available data from the literature and the Yucca Mountain program (SNL, 2007ah). For thorium, the limiting solid was ThO_2(am), the solubility model for which produced values more consistent with experimental measurements than models for the lower solubility, crystalline solid ThO_2. [In mineral formulas, "(am)" indicates that this is an amorphous, rather than orderly crystalline, solid. Amorphous solids tend to have higher solubilities than the corresponding crystalline forms.] The americium model used $AmOHCO_3$, which was identified as the controlling solid in Yucca Mountain program studies conducted under appropriate conditions. Protactinium was treated by analogy with neptunium and thorium; that is, the Np_2O_5 model was adopted for protactinium, with wide uncertainty terms accounting for the possibly lower concentrations if behavior was similar to thorium. The model for tin [addressed in SNL (2007ah) but not in the SAR] was also developed using available literature data and considerations of uncertainties; the selected controlling solid in the tin model was SnO_2(am). DOE used barium as a radium analog for solubility calculations due to their similar chemical properties. DOE constructed a pH-dependent, stepwise radium solubility limit on the basis of the model results.

NRC Staff's Review

DOE's concentration limits abstractions for the actinides thorium, americium, and protactinium are reasonable because they were based on solubility models constructed using appropriate solubility-limiting phases. The thorium model was supported by comparison with experimental solubility data that were independent of those used to develop the model. DOE supported the americium model by showing that the uncertainty range of predicted concentrations exceeded all americium concentrations measured in spent nuclear fuel dissolution experiments. Although the independent solubility data used to support the protactinium model were sparse, DOE additionally supported the model by noting that studies show protactinium solubility to be consistently lower than neptunium, which DOE used for modeling protactinium. The tin concentration limit abstraction is reasonable, because DOE based it on appropriate solubility models and corroborated the results by comparison to an independent modeling study. In addition, the choice of the tin solubility-limiting solid SnO_2(am) is reasonable because it is more soluble than the other considered tin mineral cassiterite (SnO_2).

The NRC staff notes that DOE's radium concentration limit abstraction is reasonable. The chemical analogy to barium is supported by literature observations (cited by DOE) showing their similar geochemical behaviors. DOE appropriately compared the adopted values to literature data on radium solubility, showing that the abstracted limits are similar to or higher than published model and experimental values. In addition, radium is expected to be coprecipitated in sulfates of other alkaline earth elements, particularly barium (e.g., Zhu, 2004aa), such that dissolved radium would be constrained to very low concentrations. The pure-phase radium solubility model is, therefore, reasonable and conservative.

Uncertainty

In the concentration limits abstraction, DOE addressed uncertainty in (i) thermodynamic data supporting the solubility models and (ii) the effects of fluoride ion concentration. These uncertainties are applied as additional sampled terms added or subtracted to the lookup table values, with a pH-dependent coefficient applied to the sampled fluoride uncertainty term. These thermodynamic and fluoride uncertainty terms—with normal and triangular distributions, respectively—are sampled once per realization for each element.

NRC Staff's Review

DOE appropriately accounted for uncertainty in thermodynamic constants by using sampled uncertainty terms that were based on literature thermodynamic studies DOE cited. Regarding model uncertainty, DOE explicitly accommodated the potential effects of fluoride on solubilities in uncertainty terms for the actinide abstractions. Uncertainty in chemical effects on solubility limits is applied implicitly by the in-package chemistry abstraction (TER Section 2.2.1.3.4.3.1).

Integration in the Engineered Barrier System Radionuclide Transport Abstraction

The dissolved concentration limits abstraction provides maximum concentration values for each element to the engineered barrier system radionuclide transport model, with different values provided for the waste form domain, corrosion products domain, and the invert.

NRC Staff's Review

DOE appropriately deployed the concentration limits using the GoldSim code in each domain, because the implementation accounts for the different chemical conditions. The NRC staff confirmed expected behavior by inspecting plots of modeled dissolved concentrations in the engineered barrier release domains in the TSPA analysis (e.g., DOE, 2009dc). On the basis of the NRC staff's review of the plots, solubility limits in the TSPA model were within the ranges appropriate for the given domain, as would be predicted by DOE's lookup tables and modeled chemical conditions. In addition, the radioelement concentration ranges either coincided with or were below the corresponding concentration limits.

Summary of NRC Staff's Review of Concentration Limits

The NRC staff notes that DOE's abstraction of radioelement concentration limits is appropriately integrated into the engineered barrier system radionuclide release and transport abstraction. The solubility limit models reasonably account for the engineered barrier system design, the response of system components to changing in-package conditions, and the effects of those responses on the chemical environment. DOE used equilibrium geochemical models, with reasonably chosen solubility-limiting solid phases, to construct concentration limits lookup tables. Data supporting the various element abstractions are reasonable. Model support is reasonable; DOE compared model results to appropriate laboratory data on solubility limits and concentrations during waste form dissolution studies. DOE reasonably propagated model and data uncertainty through the abstractions.

2.2.1.3.4.3.4 Availability and Effectiveness of Colloids

This section describes the NRC staff's review of DOE's abstraction and Total System Performance Assessment (TSPA) model implementation of the type, stability, and mass concentration of colloid suspensions in the engineered barrier system, as described in Safety Analysis Report (SAR) Section 2.3.7.11 and references cited therein. Colloid suspensions inside the engineered barrier system are important to repository performance, because if they are stable and exist at sufficiently high concentrations, they could enhance transport of radionuclides that are reversibly or irreversibly associated with them. (In this chapter, the term "irreversible colloids" refers to colloids with radionuclides irreversibly, or permanently, attached to them. The term "reversible colloids" refers to colloids to which radionuclides may attach and detach reversibly.)

On the basis of the NRC staff's evaluation of the TSPA code results, the NRC staff notes that the igneous intrusion and seismic ground motion modeling cases are potentially the most significant to risk with respect to long-term repository performance. The NRC staff verified that irreversible colloids are modeled in the TSPA code as independent species, separate from reversible colloids and dissolved ionic species. The NRC staff verified that radionuclides that are reversibly bound onto colloids can move back into solution and, hence, become part of dissolved radionuclide mass even if reversible colloids are unstable and settle out. Therefore, the NRC staff's review focused specifically on processes and features that limit the stability and mass concentrations of irreversible colloid suspensions [including plutonium-rich zirconium oxide (commercial spent nuclear fuel) colloids, glass waste form colloids, and iron oxide colloids] in the engineered barrier system under igneous intrusion and seismic ground motion modeling cases.

The engineered barrier system colloid model calculates the mass concentrations of reversible and irreversible colloid suspensions in the waste package, corrosion products, and invert domains of the engineered barrier system on the basis of temporal variations in aqueous chemical characteristics (pH and ionic strength), flow rates, and failure status of the engineered barrier system components under nominal and disruptive modeling cases.

Inputs to the engineered barrier system colloid mass concentration abstraction, described in SNL Section 1.1 (2008ak), include in-package ionic strength and pH from the in-package chemistry abstraction (TER Section 2.2.1.3.4.3.1), and in-drift ionic strength and pH from the engineered barrier system physical and chemical environment abstraction (TER Section 2.2.1.3.3). Mass concentrations of colloidal suspensions are used to calculate colloid-assisted radionuclide transport in the engineered barrier system radionuclide transport abstraction (TER Section 2.2.1.3.4.3.5).

Colloid Types and Radionuclides Associated With Colloids in the Engineered Barrier System

Colloids are 1 to 2-μm [4 to 8 × 10^{-5}-in]-sized particles; have the potential to facilitate transport of highly sorbing, low solubility radionuclides; and may allow radionuclide concentrations in water above their solubility limit. In the TSPA code, colloids in the engineered barrier system are formed by degradation of waste package internals and waste forms and also exist as groundwater colloids in seepage water. DOE used the engineered barrier system colloid abstraction in the TSPA code to determine the stability and mass concentrations of reversible and irreversible colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system (SAR Section 2.3.7.11).

The engineered barrier system colloid model abstraction focuses on the following five colloid suspension types: (i) glass waste form colloids, (ii) plutonium-rich zirconium oxide commercial spent nuclear fuel colloids, (iii) oxidized uranium colloids derived from the spent nuclear fuel, (iv) iron oxide colloids, and (v) groundwater colloids. The conceptual model identifies two types of radionuclide attachment to colloids (i) reversible (glass waste form colloids, oxidized uranium colloids, iron oxide colloids, and groundwater colloids), in which the radionuclides are reversibly (temporarily) sorbed onto colloid surfaces, and (ii) irreversible (glass waste form colloids, commercial spent nuclear fuel colloids, and iron oxide colloids), in which the radionuclides are permanently attached to or embedded in the colloid structure, as detailed in SNL Section 6.3.1 (2008ak) and SNL Section 6.3.4.4 (2007aj). Although glass waste form colloids, oxidized uranium colloids, iron oxide colloids, and groundwater colloids are considered in the waste form domain, iron oxide colloids are excluded from the waste form domain, as outlined in SNL Section 6.5.2.5 (2007aj) and DOE Enclosure 2 (2009ay). As shown by the data in SNL Tables 6.3.7-62, 6.3.7-63, 6.3.7-64, and 6.3.7-66 (2008ag), the sampled stable glass waste, commercial spent nuclear fuel, oxidized uranium, and groundwater colloid concentration ranges are 0.0004 to 2 mg/L [ppm], 0.000015 to 0.6 mg/L, 0.001 to 200 mg/L, and 0.001 to 200 mg/L, respectively, in all the engineered barrier system domains. As shown in SNL Table 6.3.7-65 (2008ag), the sampled stable iron oxide colloid concentration range is 0.001 to 30 mg/L [ppm] in the corrosion products and invert domains.

In the engineered barrier system colloid model abstraction, two radioelements (plutonium and americium) are modeled to permanently attach onto iron oxide colloids or be irreversibly embedded in glass waste form and commercial spent nuclear fuel colloids. As described in SNL Section 6.3.12.1 (2008ak), five radioelements (plutonium, americium, thorium, protactinium, and cesium) are modeled to reversibly sorb onto glass waste form colloids, eight radioelements (plutonium, americium, thorium, protactinium, cesium, tin, neptunium, and radium) reversibly sorb onto oxidized uranium colloids, three radioelements (plutonium, thorium, and neptunium) reversibly sorb onto iron oxide colloids, and four radioelements (uranium, neptunium, tin, and radium) reversibly sorb onto groundwater colloids. The irreversible colloids (and their associated masses of plutonium and americium) are modeled as independent species, separate from the dissolved plutonium and americium masses.

NRC Staff's Review

The NRC staff's review verified that DOE relied on laboratory and field-scale experimental results published in technical articles found in peer-reviewed journals to determine representative waste form and groundwater colloid types, and types of radionuclides reversibly and irreversibly associated with these colloids in the engineered barrier system colloid model abstraction, as detailed in SNL Section 6.3 (2008ak). DOE considered both reversible and irreversible colloids in the abstraction and noted the uncertainties associated with the mass concentrations, the stability of these colloid types, and their upscaling and applicability to the Yucca Mountain site. DOE constructed an empirical ionic strength threshold versus pH curve on the basis of existing experimental data in the open literature to account for uncertainties in the colloid stability. If the computed ionic strength for the in-package (or in-drift) environment is below the ionic strength threshold, then the colloids are stable in the corresponding environment. DOE constructed empirical cumulative (uncertainty) distributions for the mass concentration of colloids by bounding the distributions using experimental data. DOE addressed the uncertainty in the mass concentration of colloids by randomly sampling the mass concentration of stable colloids from the corresponding uncertainty distributions. The NRC staff notes that DOE provided sufficient experimental evidence from the literature and provided adequate technical justifications for the choice of colloid types and the uncertainty ranges for their mass concentrations in the engineered barrier system model abstraction.

Excluded Colloid Processes

In the engineered barrier system colloid model abstraction, colloidal filtration, thin-film straining (retardation of colloid transport when colloid dimensions exceed the water film thickness), gravitational settling of colloids, and sorption of colloids on stationary surfaces and onto an air–water interface were excluded due to associated uncertainties, and exclusion of these processes was considered to be conservative. The abstraction did not include biocolloids, because of low microbial activity and negligible mass concentrations of such colloids in comparison to groundwater colloids.

NRC Staff's Review

The NRC staff notes that DOE reasonably excluded colloidal filtration, straining, gravitational settling of colloids, and sorption of colloids on stationary surfaces and onto an air–water interface, as detailed in SNL Section 5.8 (2008ak) and SNL Section 5.7 (2007aj). Exclusion of these processes results in higher modeled radionuclide releases, and hence exclusion of these processes is conservative. Moreover, DOE's exclusion of biocolloidal (e.g., viruses, bacteria, spores, or other microorganisms) transport is reasonable because the low humic substance concentrations at the Yucca Mountain site will not support high biocolloid concentrations.

Importance of Colloids to Risk Under Disruptive Events

The NRC staff's review of the importance of colloids to risk focused on two disruptive scenario modeling cases: igneous intrusion and seismic ground motion. DOE showed, through its analyses, that these two modeling cases contribute the most to the total mean annual dose for 10,000 and 1 million years after repository closure (SAR Figure 2.4-18). These modeling cases are therefore useful for examining the potential significance to risk of colloids. Pu-242 is the most important contributor to the overall total mean dose after 200,000 years (SAR Figure 2.4-20). Under the nominal scenario, the maximum Pu-242 activity due to irreversible colloids is about 30 percent of the total Pu-242 activity leaving the engineered barrier system; under the seismic scenario, it is only about 18 percent of the total Pu-242 activity leaving the engineered barrier system. The maximum Pu-242 release rate leaving the engineered barrier system due to irreversible colloids is 2.5 percent of the total release rate under the igneous intrusion modeling case, as described in SNL Section P18.3 (2008ag) and DOE Enclosure 5 (2009an). DOE concluded that, for plutonium in the engineered barrier system, colloid-facilitated radionuclide transport is less effective than dissolved phase radionuclide transport, as shown in SAR Figures 2.1-20 and 2.1-23 and SNL Table A–2, p. A–130 (2008ad).

In DOE's igneous intrusion modeling case, the drip shield and waste packages entirely fail; hence there is no distinction between seep and no-seep cases, and water chemistries and colloid stability remain nearly constant once the temperature drops below the boiling point and water flow to the waste form is established. For the igneous intrusion modeling case, smectite colloids (derived from high-level waste glass or from the tuff host rock) and oxidized uranium colloids (derived from spent nuclear fuel degradation) are stable in the engineered barrier system, but commercial spent nuclear fuel colloids are completely unstable in the corrosion products domain, according to DOE Enclosure 3 (2009ay).

For the igneous intrusion case, unstable and settled commercial spent nuclear fuel colloids in the corrosion products domain can be as important as the stationary corrosion products for the retention of Pu-242, as described in DOE Figure 1.1-26 (2009dc), DOE Enclosure 1 (2009dd), and DOE Enclosure 3 Figure 1 (2009ay), due to a narrow range of pH, 7 to 8.4. The narrow pH range is illustrated in DOE Enclosure 3 (2009ay) and supported in DOE Enclosure 2 (2009da). The contribution of suspended iron oxide colloids to Pu-242 mass in the corrosion products domain is about eight orders of magnitude smaller than the Pu-242 mass removed from inventory by the settled unstable commercial spent nuclear fuel colloids at 200,000 years for the igneous intrusion modeling case, as shown by a representative realization in DOE Figure 1.1-26 (2009dc). Therefore, DOE concluded that iron oxide colloids are insignificant for plutonium mobility when commercial spent nuclear fuel colloids sequester Pu-242. In contrast, DOE Enclosure 1 (2009da) shows that, in 62 percent of realizations in the igneous intrusion case, commercial spent nuclear fuel colloids are unstable in the waste form domain, but iron oxide colloids are stable in the corrosion products domain. Therefore, DOE concluded that commercial spent nuclear fuel colloids have a negligible effect on Pu-242 waste package mobility in these cases. For a representative realization for these conditions, Pu-242 mass irreversibly associated with iron oxide colloids is three orders of magnitude lower than the Pu-242 mass in the dissolved phase and about seven orders of magnitude lower than the sorbed Pu-242 mass on stationary corrosion products, as shown in DOE Enclosure 1 Figure 7 (2009da). DOE stated that this observation indicates insignificant effects of iron oxide colloids on Pu-242 waste package releases.

For the seismic ground motion case, DOE assessed that damage on waste packages is mainly due to patch failures by general corrosion. Unlike the igneous intrusion modeling case, colloid concentrations are sensitive to seep versus no-seep environments after corrosion patches develop on the waste packages, and the ionic strengths of water and pH vary in time. For the seismic ground motion case, the ionic strength of waters in the engineered barrier system depends on the relative humidity when the water flux is less than 0.1 L/yr [0.026 gal/yr] under the condition of complete filling of the drift with rubble; otherwise, ionic strength depends on the chemistry of the advective flux through corrosion patches. The ionic strengths of seep water also correlate with the rubble-filling status, as shown in DOE Enclosure 3, Figures 17, 18, 23, and 24 (2009ay).

For the seismic ground motion modeling case, DOE noted that diffusive transport (under no-seep conditions) through the engineered barrier system and the water chemistry (pH and ionic strength) could largely limit colloid-facilitated transport, as shown in Safety Analysis Report (SAR) Section 2.4.2.2.3.2.2 and DOE Enclosure 3 (2009ay).

The Total System Performance Assessment (TSPA) model results indicate that for the seismic ground motion case, initially high ionic strength leads to unstable colloid suspensions. After the waste packages are breached in both seep and no-seep conditions, the ionic strength (which depends on the relative humidity during this stage) drops to a level where smectite and oxidized uranium colloids become stable in the codisposal packages and largely stable (in more than 95 percent of realizations) in the commercial spent nuclear fuel waste packages. These colloids are stable for the remainder of the simulation. As was the case for the igneous intrusion modeling case, commercial spent nuclear fuel colloids are unstable in the corrosion products and invert domains for the seismic ground motion modeling case, as shown in DOE Enclosure 3, Figures 6, 29, and 30 (2009ay). Stability of colloid suspensions, except for the iron oxide colloids, is similar for both the seep and no-seep cases. Iron oxide colloids are stable in the corrosion products domain only when seep water enters the waste package through corrosion patches at later times, whereas they are largely unstable (in more than 95 percent of realizations) under the no-seep conditions, as shown in DOE Enclosure 3, Figures 15 and 16 (2009ay).

NRC Staff's Review

DOE reasonably concluded that iron oxide colloids are insignificant contributors to Pu-242 releases from the corrosion products domain, because Pu-242 is either (i) largely sorbed onto stationary corrosion products, (ii) associated with settled and unstable commercial spent nuclear fuel colloids, or (iii) in the dissolved phase.

The NRC staff reviewed the TSPA model results for the disruptive modeling cases (highly significant to risk) to evaluate processes and features that could limit availability and transport of colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system, as detailed in DOE Enclosure 3 (2009ay). The NRC staff notes that DOE appropriately identified, described, and quantified the distinct processes and features that control stable colloid concentrations in the engineered barrier system (e.g., considering barrier capability of engineered barrier system components, the impact of seep and no-seep environments) for both the igneous intrusion and the seismic ground motion modeling cases. In addition to the simpler, long-term constant geochemical conditions considered in the igneous modeling case, the seismic ground motion modeling case demonstrated the effects of temporal variability in the stability and mass concentrations of colloids under both seep and no-seep conditions, as a function of patch failure developments on waste packages and resulted changes in ionic strength, pH, and relative humidity in the engineered barrier system.

The NRC staff conducted independent, simplified, confirmatory calculations on the effectiveness of iron oxide colloids in facilitating Pu-242 releases in the igneous intrusion modeling case. The igneous intrusion modeling case was chosen for independent calculations because (i) chemical conditions and colloid stability remain unchanged throughout the entire simulation after a relatively short cooling period (less than 1,000 years), (ii) this modeling case dominates the long-term total mean annual dose, and (iii) that dose is dominated by Pu-242 after 200,000 years. Using information provided by DOE and used for Pu-242 in the TSPA model, the NRC staff's confirmatory calculations show that the ratio of (i) the plutonium attachment rate to iron oxide colloids to (ii) the plutonium attachment rate to stationary corrosion products is 2×10^{-7} (Pickett, 2010aa). These calculations showed that plutonium attachment to stationary corrosion products is much faster than attachment to iron oxide colloids.

As discussed in Technical Evaluation Report (TER) Section 2.2.1.3.4.3.5, DOE showed in DOE Enclosure 8 (2009ay) that reversible, kinetic plutonium sorption onto stationary corrosion products can be approximated as an equilibrium process. Therefore, the rate of plutonium desorption from the stationary corrosion products is approximately equal to the rate of sorption. On the basis of this observation and the NRC staff's confirmatory calculation summarized in the previous paragraph, the NRC staff notes that the rate of irreversible plutonium sorption to iron oxide colloids is many orders of magnitude slower than the rate of plutonium desorption from stationary corrosion products to solution. Therefore, any transfer of dissolved plutonium to iron oxide colloids would be instantaneously compensated by desorption from the stationary corrosion products—which contain the majority of plutonium mass in the corrosion products domain—to maintain the quasi-equilibrium relationship. On the basis of its calculation, the NRC staff notes that irreversible sorption of plutonium to iron oxide colloids cannot substantially deplete dissolved plutonium. Sorption to stationary corrosion products is more important to plutonium release from the corrosion products domain than are iron oxide colloids. This result is consistent with DOE's conclusion that iron oxide colloids are not significant for Pu-242 releases from the engineered barrier system, as shown for representative realizations in DOE Figures 1.1-24 and 1.1-26 (2009dc) and DOE Enclosure 1, Figures 5 and 7 (2009da).

For codisposal packages in disruptive scenarios, the NRC staff compared the plutonium release effectiveness of high-level waste glass colloids against dissolved plutonium. Results of a representative igneous intrusion modeling case realization in DOE Enclosure 1, Figures 1.1-29 and 1.1-30 (2009dc) showed dissolved Pu-242 being released from the engineered barrier system more than 10 times faster than Pu-242 associated with high-level waste glass colloids. In addition, the mean plutonium solubility limit in the corrosion products domain in the TSPA model, which the NRC staff estimated as 10^{-5} g/L [0.01 ppm] from DOE data presented in SNL Table 6.5-1 (2007ah), is about 10 times higher than the mean concentration of plutonium associated with high-level waste glass colloids, which the NRC staff calculated as 1.2×10^{-6} g/l [0.0012 ppm] from data presented in SNL Table 6-4 (2008ak). The NRC staff notes, on the basis of these confirmatory calculations, that high-level waste glass colloids are, in general, less effective than the dissolved phase in effecting plutonium release from codisposal packages.

The NRC staff considered (i) the insignificance of iron oxide colloids for Pu-242 releases from the corrosion products domain; (ii) the relatively shorter residence times for colloids and radionuclides in the waste form domain than in the corrosion products domain; and (iii) the presence of nearly 60–70 percent less steel corrosion product mass in the waste form domain than in the corrosion products domain, as described in DOE Enclosure 2 (2009ay), and notes that excluding iron oxide colloids from the waste form domain will not be significant to risk.

Reduction in colloid mass concentration under no-seep environments (which leads to diffusion-dominant colloid migration) is reasonable, because the size of the colloids is a few orders of magnitude larger than the ions of aqueous radioelements, and the colloids therefore will encounter higher diffusional resistance. Colloids diffuse much more slowly than dissolved ionic species, such that colloid-associated release under no-seep conditions is negligible.

In summary, for the reasons identified previously, DOE's analysis of colloids under disruptive igneous and seismic events is reasonable.

Data Support and Uncertainty Propagation for Colloid Transport Abstraction for the Engineered Barrier System

Mass Concentration of Colloids

DOE used experimental and scientific literature data to support its colloid mass concentrations model. Uncertainties associated with the mass concentrations and stability of glass waste form colloids in the TSPA model rely on results from drip and immersion tests for degradation of alkali borosilicate glasses, as detailed in SNL Section 6.3.2.2 (2008ak). Experimental data were used to bound plutonium mass concentrations associated with zirconium oxide colloids formed from commercial spent nuclear fuel, as described in SNL Section 6.3.2.4 (2008ak). Uranophane colloids are used as representative colloids for oxidized uranium colloids formed from defense and commercial spent nuclear fuel.

SNL Section 6.3.2.6 (2008ak) described an empirical cumulative distribution for the mass concentrations of groundwater colloids that was adopted for uranophane colloid suspensions, because both colloid suspensions display a similar stability profile. DOE relied on bench-scale experiments using a carbon-steel miniature waste package in bathtub and flow-through configurations. These experiments used water chemically similar to well water near Yucca Mountain to induce corrosion and subsequently to determine the geometric mean concentration of iron oxide colloids. Empirical cumulative distributions of colloid mass concentrations were constructed, on the basis of laboratory-scale experimental data, to address uncertainties in colloid mass concentrations, as detailed in SNL Section 6.3 (2008ak). DOE adopted colloid concentrations in groundwater at the Yucca Mountain site for colloid concentrations in seepage water entering breached waste packages. DOE collected colloid data from nine different sources and fitted them to a cumulative mass distribution to address uncertainties.

NRC Staff's Review

With regard to DOE's colloidal mass concentration model, the NRC staff verified that DOE relied on data from laboratory experiments published in peer-reviewed technical journals to determine the range for mass concentrations of reversible and irreversible colloids. The NRC staff reviewed (i) how DOE addressed uncertainties associated with how closely geochemical and hydrogeological conditions were represented in these experiments and (ii) how DOE upscaled laboratory results to the field scale at the Yucca Mountain site. DOE acknowledged uncertainties associated with the mass concentrations of colloid suspensions due to, among other things, measurements, experimental factors, and upscaling of experimental data and observations to repository scale and conditions (SAR Section 2.3.7.11.2). To be conservative, DOE set the upper bound for mass concentrations of iron oxide colloids to be larger than natural iron oxide colloid concentrations in groundwater (SAR Section 2.3.7.11.2). The NRC staff notes that, in the absence of field data, the use of laboratory data is reasonable, because DOE incorporated associated uncertainties (through sampling from uncertainty distributions) in determining the mass concentration of waste form and iron oxide colloids. For mass concentrations of groundwater colloids, DOE appropriately employed existing field data. DOE kept the pH and ionic strength range, which was based on experimental and/or literature data, wide enough in colloid stability analyses to cover expected stability conditions at the Yucca Mountain site.

The NRC staff's review verified that DOE addressed uncertainties in mass concentrations by sampling them from empirically constructed cumulative mass distributions obtained from experimental data. DOE (i) adequately propagated data uncertainty by constructing empirical mass concentration distribution functions for each colloid type using relevant experimental data and sampling the mass concentration for a particular colloid suspension type and (ii) adequately addressed model integration and information exchange between the engineered barrier system abstraction and other abstractions.

In-Package and In-Drift Stability of Colloids

In Total System Performance Assessment (TSPA) code calculations, in-package and in-drift stability of colloid suspensions is determined by the ionic strength of the seepage water and pH. DOE constructed an empirical ionic strength threshold versus pH curve using experimental data specific to each colloid suspension type and using the Derjaguin-Landau-Verwey-Overbeek theory. In-package and in-drift pH and ionic strengths and dissolved radionuclide concentrations are computed outside the engineered barrier system abstraction and then fed into the empirical ionic strength threshold versus pH curve in the engineered barrier system abstraction, as outlined in SNL Section 6.5 (2008ak). For stability calculations, DOE modeled glass waste form colloids and groundwater colloids as smectite, plutonium-rich zirconium oxide colloids as zirconium oxide, oxidized uranium oxide colloids as uranophane, and iron oxide colloids as hematite, as described in SNL Section 6.3.1 (2008ak). If the colloid suspensions are stable, their mass concentrations are sampled from empirical distribution functions specific to the colloid type (constructed from experimental data). If a colloid type is unstable, the mass concentration is set to a low nonzero value, selected such that the colloid mass is too low to have any impact on radionuclide release and transport. In the case of groundwater colloids in the waste package, the initial concentration is set to 0 mg/L until flow begins in the waste package (SNL, 2008ak).

NRC Staff's Review

The NRC staff's review of the in-package and in-drift colloidal stability verified that DOE constructed empirical relations (on the basis of experimental data in the literature) for each colloid suspension type that related the ionic strength threshold to pH to determine stability of colloidal suspension in the waste package and in the drift. These empirical relations were constructed on the basis of the Derjaguin-Landau-Verwey-Overbeek model. DOE acknowledged that these empirical relations were purely mathematical fits and had no physical meaning, as stated in SNL Section 6.3.2.4 (2008ak), although they are driven by experimental data. The engineered barrier system model abstraction computes the ionic strength of the in-package fluid (or in-drift seepage fluid) and compares it against the ionic strength threshold read from the ionic strength threshold versus pH curve. If the in-drift (or in-package) fluid ionic strength exceeds the ionic strength threshold, then the corresponding colloidal suspensions become unstable in the abstraction, as detailed in SNL Section 6.3.2 (2008ak). The stability calculations for colloid suspensions are reasonable, because (i) the Derjaguin-Landau-Verwey-Overbeek theory has been commonly used in the literature to determine stability of relatively dilute colloid suspensions [DOE provided references to support its common use in SNL Section 6.3.2 (2008ak)], (ii) the empirical relations for ionic strength versus pH were constructed on the basis of experimental data, and (iii) DOE addressed temporal variations in in-package and in-drift water chemistry in colloid stability calculations.

Radionuclide Mass Sorption on Colloid Suspension

DOE referred to previously published data to determine surface area for reversible glass waste and groundwater colloids, uranophane colloids, and iron oxide colloids in SNL

Sections 6.3.2.3.1, 6.3.2.7, and 6.3.12.2 (2008ak), respectively. This information is used in calculating sorbed radionuclide mass on colloid suspensions.

NRC Staff's Review

The NRC staff's review verified that DOE determined the range of specific surface area for reversible colloid suspension based on open literature experimental data and sampled from this range to address data uncertainties. On the basis of the NRC staff's confirmatory review of the literature data cited in SNL (2008ak) and DOE's use of sampled specific surface area to account for uncertainties, the NRC staff notes that the uncertainty distributions and literature data DOE provided for specific surface area are reasonable.

Kinetic Attachment Rates for Plutonium and Americium Onto Iron Oxide Colloids

In the engineered barrier system abstraction, plutonium and americium are modeled to be irreversibly attached onto iron oxide colloids. As described in SNL Section 6.3.12.2 (2008ak), DOE constructed an uncertainty distribution function for the attachment rate constant for iron oxide colloids on the basis of data from sorption experiments. DOE noted that the attachment rate is sampled from an experimentally supported lognormal uncertainty distribution under a no-seep condition or a condition where colloid suspensions are unstable in the corrosion products domain. Otherwise, the maximum of the sampled rate constant from a lognormal uncertainty distribution and the computed rate constant using the target flux out ratio are used, as detailed in SNL Section 6.5.2.4.6 (2007aj).

NRC Staff's Review

The NRC staff reviewed the basis of DOE's description for the attachment rate calculations and notes that DOE used attachment rates sampled from an experimentally supported uncertainty distribution for modeling irreversible attachments of plutonium and americium onto iron oxide colloids. The model favors attachment onto iron oxide colloids by implementing the target flux out ratio if the computed attachment rate remains within the experimentally determined range for the attachment rate; otherwise, the sampled attachment rate is used and the target flux out ratio is not implemented. This modeling feature is conservative, because it would allow more radionuclides to be transported from the corrosion products domain to the invert by stable iron oxide colloids (TER Section 2.2.1.3.4.5). The NRC staff further notes that because iron oxide colloids do not play a significant role in dose calculations (as elaborated in the previous section), the method chosen for irreversible attachment rate calculations is not important to dose calculations.

Alternative Conceptual Model Consideration

DOE considered two alternative conceptual models with respect to colloids: the first uses different mechanisms to generate glass colloids and the second focuses on air–water limitations on the releases of particles from weathered waste form surfaces under unsaturated conditions, as identified in SAR Section 2.3.7.11.3.2 and SNL Section 6.4 (2008ak). DOE did not implement these alternative models in the TSPA.

NRC Staff's Review

The NRC staff's review verified that DOE used a conceptual model in the TSPA for irreversible and reversible colloidal transport on the basis of a set of mass-balance equations. The NRC staff notes that a mathematical framework for this conceptual model is consistent with colloid transport models in the literature (e.g., Corapcioglu and Jiang, 1993aa; van de Weerd, et al., 1998aa). DOE does not have sufficient experimental data to fully evaluate the aforementioned two alternative conceptual models. It is reasonable that the two alternative conceptual models were excluded from TSPA, because the technical basis for including and quantifying the models is insufficient for use in a performance assessment calculation.

DOE considered the bathtub flow model as an alternative conceptual model to the flow-through model, which is implemented in TSPA, in simulating water flow and radionuclide transport in a breached waste package. As discussed in Technical Evaluation Report (TER) Sections 2.2.1.3.3.3.3 and 2.2.1.3.4.3.5, the flow-through model is bounding to the bathtub model for flow and radionuclide transport simulations. DOE did not discuss potential implications of the bathtub flow model on the colloid transport in the engineered barrier system. This was reasonable, because if the commercial spent nuclear fuel colloids are unstable in the corrosion products domain during pulse periods, large fractions of commercial spent nuclear fuel colloids irreversibly associated with Pu-242 would be removed from the inventory, and hence, the implementation of bathtub model in TSPA would underestimate Pu-242 releases from the engineered barrier system.

Summary of NRC Staff's Review of Availability and Effectiveness of Colloids

DOE adequately described the engineered barrier system colloid transport model abstraction and its integration with other abstractions used in the TSPA code. The mathematical framework and the underlying conceptualization for the reversible and irreversible colloids and colloid stability analyses are consistent with models in the scientific literature. The absence of other conceptualizations used in the TSPA code is reasonable, given that sufficient data were not available to support alternative models. The abstraction appropriately propagates uncertainties through laboratory-data- or field-data-based cumulative distributions for mass concentrations, stability, and transport parameters.

DOE used disruptive modeling cases that are most significant to risk (igneous intrusion and seismic ground motion modeling cases) to show the ineffectiveness of colloid-assisted radionuclide releases from the engineered barrier system in comparison to radionuclide releases in a dissolved phase. On the basis of the NRC staff's evaluation of the TSPA code results and the NRC staff's independent confirmatory analyses, the NRC staff notes that the limitation on stable colloid masses is the main reason for the insignificance of colloid-assisted radionuclide transport in both seep and no-seep environments. Moreover, diffusive release by colloids is limited due to size effects, making colloid transport further ineffective under no-seep environments. The NRC staff notes that dissolved radionuclides will be more significant than colloid-associated radionuclides to radionuclide release and transport and, therefore, to dose.

2.2.1.3.4.3.5 Engineered Barrier System Radionuclide Transport

This section details the NRC staff's review of DOE's abstraction and Total System Performance Assessment (TSPA) implementation for radionuclide transport in the engineered barrier system, as described in Safety Analysis Report (SAR) Section 2.3.7.12 and references cited therein (particularly SNL, 2007aj). The abstraction estimates the rate of movement of radionuclides from degraded waste forms to the unsaturated zone and provides radionuclide fluxes (rates of mass transfer) versus time to the unsaturated zone transport abstraction (TER Section 2.2.1.3.7). Major inputs to the abstraction include the flow conditions inside the engineered barrier system (TER Section 2.2.1.3.3.3.3), the chemical conditions inside the engineered barrier system (TER Section 2.2.1.3.4.3.1), waste form degradation rates (TER Section 2.2.1.3.4.3.2), dissolved concentration limits (TER Section 2.2.1.3.4.3.3), and colloid parameters (TER Section 2.2.1.3.4.3.4).

DOE's abstraction for radionuclide transport in the engineered barrier system is highly significant to risk because large masses of plutonium and other dose-significant actinides are retained in the engineered barrier system in DOE's TSPA calculations. For example, in DOE (2009dc), DOE provided results for a representative realization of the igneous intrusion modeling case showing that approximately 8,000 kg [17,600 lb] of Pu-242 is permanently immobilized in the engineered barrier system for one percolation subregion. In the same realization and subregion, approximately 30,000 kg [66,000 lb] of Np-237 is retained on the waste package corrosion products at 100,000 years; Np-237 is released from the engineered barrier system slowly enough that more than 1,000 kg [2,200 lb] remained at 1 million years.

On the basis of the importance to the abstraction, the NRC staff's review focused on model framework and process conceptualization within the TSPA code implementation, representation of diffusion, sorption on stationary corrosion products, colloid-facilitated transport, and reasonableness and consistency of TSPA code results. The abstraction contains several included features, events, and processes. Excluded features, events, and processes are discussed in TER Section 2.2.1.2.1.3.2.

Model Framework and Process Conceptualization

Overall Conceptualization

DOE based the abstraction and TSPA implementation on one-dimensional mass transport through three computational domains (i) waste form domain, (ii) corrosion products domain, and (iii) invert domain. The waste form domain contains a single computational cell representing a porous rind of degraded waste form for the commercial spent nuclear fuel packages. The waste form domain for codisposal packages comprises a computational cell representing high-level waste glass upstream of a cell representing DOE spent nuclear fuel. Corrosion products formed from the degradation of steel waste packages and package internals are represented in the corrosion products domain. The invert domain is assumed to be in close contact to the waste package and composed of crushed tuff material. A fourth domain, the invert-unsaturated zone interface, facilitates transfer of the radionuclide mass from the engineered barrier system transport model to the unsaturated zone transport model.

DOE conceptualized the transport pathway as a flow-through model in which water flows vertically through a degraded waste package. DOE considered an alternative conceptual model in which the outlet for water is not on the underside of the waste package. In this bathtub model, water would fill the partially intact waste package until it reaches a spill point corresponding to a breach on the side of the waste package. In a variant of the bathtub model, the stored water and dissolved radionuclides would be suddenly released when a second breach develops on the underside of the waste package.

NRC Staff's Review

The major structure of the transport abstraction, which is based on a one-dimensional spatial discretization, reasonably represents the transport pathway in the engineered barrier system with sufficient flexibility to represent the range of expected conditions and processes.

DOE has shown that the flow-through model provides an upper bound on radionuclide transport in the absence of a sudden release based on considerations of water residence times, as discussed in SNL Section 6.6.1.2.3 (2007aj). A sudden release of water stored in the waste package in the bathtub scenario could create a short-duration, high-intensity pulse in radionuclide release from the engineered barrier system. However, several mechanisms mitigate the effects of such a pulse by dispersing it in time. The pulse from a single package would be dispersed by physical processes such as sorption and dispersion in the engineered barrier system and the lower geological barrier. Radionuclide concentrations in the pulsed water would, therefore, become lower as the water moves from the engineered barrier system and through the natural barriers. Moreover, the pulses from individual realizations would be distributed in time, and the effect of any one pulse on the mean dose would be greatly reduced because the combined, averaged effects of time-distributed pulses would be similar to the effect of continuous flow-through releases. On the basis of its review of this information, DOE's choice of the flow-through conceptualization over the bathtub model is reasonable.

Transport Model Framework

In DOE's abstraction, dissolved radionuclides and radionuclides sorbed onto the five types of mobile colloids described in TER Section 2.2.1.3.4.3.4 are transported by diffusion and, if water is flowing in the engineered barrier system, advection. Advective velocities for colloids are identical to the water velocity. Advective transport of selected dissolved radionuclides is slowed by sorption onto stationary corrosion products. Solubility limits on the dissolved radionuclide concentrations are also imposed.

NRC Staff's Review

On the basis of understanding developed during previous interactions (e.g., MacKinnon, 2008aa) and the NRC staff's experience with modeling similar systems, the NRC staff considers the represented diffusive and advective processes to be the dominant processes for transport. The NRC staff is unaware of other processes that might be expected to significantly contribute to transport. Thus, the processes and couplings represented in the abstraction are reasonably complete and do not underestimate the release of radionuclides.

Transport Under Disruptive Events

DOE's TSPA implementation of engineered barrier system transport is similar for the disruptive and nominal modeling cases, although conditions within the engineered barrier system (input for the abstraction) may be different following disruptive events. Most importantly, DOE assumed instant degradation of the waste forms and advective conditions within the engineered barrier system following an igneous intrusion event.

NRC Staff's Review

DOE's model reasonably assumed that under an igneous intrusion event, the waste forms are instantaneously degraded and are transported under advective conditions. These assumptions for igneous intrusion events provide an upper bound on engineered barrier system transport.

Model Abstraction and TSPA Model Results

In response to the NRC staff's questions regarding consistency between the model abstraction and the TSPA code calculated results, DOE provided additional information on the mass retained in and flux out of each computational domain for key representative radionuclides using single representative realizations of the igneous intrusion and nominal modeling cases (DOE, 2009da,dc). The NRC staff used the results for commercial spent nuclear fuel and the igneous intrusion modeling case to evaluate consistency between the results of the TSPA analyses and the conceptual process models DOE described. For I-129, 99.96 percent of the initial inventory is transported out of the engineered barrier system in the first TSPA timestep following the intrusion event. Release of Np-237 is significantly delayed but not eliminated by precipitation and sorption onto stationary corrosion products. For example, about 14 percent of the initial Np-237 inventory is released from the engineered barrier system in the first 40,500 years following the igneous intrusion event; after 1 million years the released fraction is 88 percent of the initial inventory (including ingrowth).

For Pu-242, DOE provided information for two realizations: one with stable waste form colloids in the waste form domain, described in DOE (2009dc), and one with unstable waste form colloids in the waste form domain, described in DOE Enclosure 1 (2009da). For the realization with stable waste form colloids in the waste form domain, which according to DOE Enclosure 1 (2009dd) is representative of about 38 percent of realizations, about 11 percent of the initial inventory is released from the engineered barrier system in the first 204,000 years. Nearly all of the remaining Pu-242 mass is retained in the corrosion products domain irreversibly associated with permanently immobilized (settled) waste form colloids in this realization. In DOE's realization results with unstable waste form colloids in the waste form domain, which is representative of 62 percent of realizations, precipitation of plutonium-bearing minerals and sorption onto stationary corrosion products delay release but do not permanently sequester Pu-242, similar to Np-237.

NRC Staff's Review

The NRC staff notes that rapid transport of I-129 following the igneous intrusion event is consistent with the conceptual process model for highly soluble, nonsorbing species, for which no significant engineered barrier system retention processes are represented. The NRC staff notes that the analyzed TSPA abstraction results for Np-237 release are consistent with the conceptual process models because dissolved neptunium concentrations are limited by precipitation of neptunium-bearing minerals and by strong sorption onto stationary corrosion products in this realization. The NRC staff notes that the mechanism by which Pu-242 is retained in the corrosion products domain is consistent with the process conceptualization when waste form colloids are calculated to be stable in the waste form domain and unstable in the corrosion products domain. Pu-242 behavior, when waste form colloids are unstable in the waste form domain, is also consistent with the process conceptualization.

Summary of Diffusion Models

The various analytical models used to simulate diffusive transport in the Total System Performance Assessment (TSPA) computer code are summarized next.

DOE calculated diffusion coefficients for dissolved radionuclides as the product of tortuosity (the effect of flow path shape in a porous medium) and species-dependent free-water diffusion coefficients. The diffusion coefficients were adjusted for temperature. The tortuosity was empirically related to porosity and liquid saturation using standard models. DOE based diffusion coefficients for colloids on the Stokes-Einstein equation, which accounts for temperature and particle size. The mathematical representation of diffusion and the approach for relating diffusion coefficients to porosity, water content, and temperature are reasonable, in that they are based on established models in the peer-reviewed scientific literature and are applied to conditions within their valid range.

For the no-dripping situation, DOE calculated liquid water content from relative humidity using empirical adsorption isotherms. This information is needed to establish diffusion coefficients, which are dependent on water content. DOE developed a diffusion model for these conditions and conducted a literature review on data relevant to predicting water content on the basis of relative humidity, as described in SNL Section 6.3.4.3 (2007aj). DOE compared the output of the abstraction for adsorbed water content versus relative humidity with literature data for adsorption on goethite, hematite, Cr_2O_3, and NiO. DOE showed that, within uncertainty bounds, the output of the abstraction is consistent with the experimental data for relative humidity of about 0.42 and greater, which is the range of relative humidity in which diffusion may be significant, as outlined in SNL p. 7-22 (2007aj). DOE also compared the model output to the results of independent modeling studies. The NRC staff reviewed DOE’s data, model development, and model corroboration and notes that the water adsorption isotherm model for diffusion under no-dripping conditions is reasonable because it was derived using applicable literature data and was corroborated.

The mass of steel corrosion products is needed to establish the liquid water content, which DOE calculated as a function of time from the degradation of steel internals of the waste package. The NRC staff compared the corrosion rates for stainless and carbon steel used in this abstraction with literature values and additional information DOE provided in DOE Enclosure 2 (2009db) and noted that the uncertainty distributions are reasonable. Stainless steel corrosion rates are appropriate for representing diffusion because a possible overestimation of the corrosion rate at the lower end of the range would be conservative for simulating diffusion. The upper end of the corrosion rate is consistent with the data and would not result in an underestimate of liquid water content (and, therefore, diffusion coefficient). The NRC staff also reviewed DOE’s use of design information in establishing the corrosion product mass and notes that the appropriate information on waste package design was used.

For the dripping situation, DOE assumed the porous materials were saturated with liquid water. This assumption is reasonable because it provides an upper bound with respect to diffusive transport; in general, contaminants will diffuse faster in a fully saturated medium than in a partially saturated medium.

DOE used data on diffusion in crushed tuff material to develop uncertainty distributions for invert diffusion coefficients. Uncertainty in the diffusion coefficients and invert porosity are explicitly represented. The NRC staff notes that the supporting experiments are representative of the expected conditions in the invert and that DOE’s uncertainty distributions are thus appropriately developed.

DOE compared the output of the invert diffusion coefficient model with two sets of experimental results. DOE’s abstraction predicts larger diffusion coefficients than the experimentally determined values. The abstraction is reasonable because it produces an upper bound of the diffusion coefficient and the diffusive transport of radionuclides in the invert.

DOE considered two alternative conceptual models related to diffusion in the invert. One of these alternative conceptual models is based on a dual-continuum representation of diffusion. DOE's selection of the single-continuum model over the dual-continuum model is reasonable because little intergranular moisture is expected in the invert except in the dripping situation, in which case diffusion is expected to be minor compared with advection. (Little moisture would imply negligible intergranular diffusion, and the adopted single-continuum model provides an upper bound on diffusive transport.) DOE's other alternative conceptual model considers alternative relationships between diffusion coefficients and moisture content at low moisture content. The NRC staff notes that, given significant gaps in the data that would support this alternative conceptual model, DOE reasonably did not adopt this model and adopted a relationship that overestimates diffusion coefficients at low water content and is consistent with available data.

In DOE's TSPA model, radionuclide mass enters the unsaturated zone after leaving the invert. Because the mass flux out of the invert is partly the result of diffusion, radionuclide concentrations in the unsaturated zone are needed to obtain estimates of the mass flux. DOE modeled a portion of the unsaturated zone in the engineered barrier system/unsaturated zone interface to calculate the diffusive fluxes into the unsaturated zone transport model. The engineered barrier system/unsaturated zone interface is a network of computational cells representing a local region of the unsaturated zone just below a drift. Hydrological conditions in the engineered barrier system/unsaturated zone interface are established similarly to the unsaturated zone transport model (TER Section 2.2.1.3.7). A zero concentration boundary condition is used at the lower boundary of the interface zone. Because of this assumption, DOE's use of an engineered barrier system/unsaturated zone interface zone to couple the engineered barrier system and unsaturated zone abstractions produces an upper bound on diffusive transport into the unsaturated zone, and therefore use of the interface zone is reasonable.

Sorption on Corrosion Products

In DOE's abstraction, radionuclides enter the corrosion products domain from the waste form domain both in solution and associated with montmorillonite, plutonium, zirconium, and uranophane colloids. The transport of selected radionuclides in the abstraction is retarded by sorption onto stationary corrosion products (SAR Section 2.3.7.12). DOE treated the corrosion products domain as a single mixing cell containing a homogenous porous medium with no preferential flow paths, on the basis of a conceptualization of degraded waste form disseminated within a corrosion product mass. The NRC staff notes that DOE's conceptualization and TSPA implementation as a mixing cell provide reasonable representations of radionuclide release because a disseminated radionuclide source supports the underlying assumption of uniform radionuclide concentrations within the represented volume.

In DOE's abstraction, corrosion product surface area is used to calculate the volume of adsorbed water and the mass of radionuclides sorbed onto corrosion products. DOE assumed a mixture of hydrous ferric oxide and goethite for calculating corrosion product surface area without considering the aging of hydrous ferric oxide to more crystalline iron oxides with lower surface area. DOE summarized additional sensitivity information in DOE Enclosure 6 (2009ay) showing that the result of the abstraction is not significantly affected by the assumed relative abundance of hydrous ferric oxide. NRC staff reviewed the sensitivity analysis and notes that the abstraction appropriately represents uncertainty in the corrosion product surface area.

The abstraction calculates the corrosion product surface area as a function of time using an uncertainty distribution of stainless steel corrosion rates based on literature data. On the basis of additional information DOE provided in BSC (2004ae) and DOE Enclosure 2 (2009db) and an independent literature review, the range of stainless steel corrosion rates that were used to simulate growth of the corrosion products that provide the sorption substrate is reasonable because it is consistent with literature corrosion rate data (Beavers and Durr, 1990aa; Glass, et al., 1984aa; McCright, et al., 1987aa).

DOE modeled sorption on corrosion products for uranium, neptunium, thorium, americium, and plutonium using a surface complexation model to develop effective distribution coefficients (K_ds) taking into account the chemical conditions. A surface complexation model simulates equilibrium attachment of dissolved ions onto solid surfaces and incorporates the chemical complexity of the system. Other radioelements that are tracked in the unsaturated zone transport abstraction (TER Section 2.2.1.3.7)—such as cesium, protactinium, radium, selenium, strontium, tin, technetium, iodine, chlorine, and carbon—are assumed in DOE's engineered barrier system transport abstraction to not sorb onto stationary corrosion products. Nickel is included in the surface complexation model to represent competition for sorption sites, but the sorbed mass of nickel is not explicitly tracked for transport purposes.

The surface complexation model is not directly incorporated in DOE's TSPA model abstraction, but the distribution coefficients developed from the surface complexation model are directly applicable to the transport of uranium, neptunium, and thorium, which undergo rapid and reversible sorption (SNL, 2007aj). Kinetic reversible sorption is modeled for americium and plutonium sorption on stationary corrosion products. The forward sorption rate constant in this case is sampled from an uncertainty distribution. The desorption rate constants are then calculated from this forward rate and the K_ds calculated using the surface complexation model.

To develop the K_d distributions used in the TSPA model abstraction, DOE carried out the surface complexation models using the PHREEQC geochemical software the U.S. Geological Survey developed. To represent parameter uncertainty in the surface complexation model, DOE conducted about 5,000 PHREEQC simulations, each with a unique combination of surface properties and aqueous chemistry parameters as inputs. DOE then analyzed the PHREEQC simulation results using multiple regressions to produce functions that calculated actinide sorption as a function of key geochemical properties. These functions provide the K_d values used in the TSPA model abstraction of sorption to stationary corrosion products. DOE's approach is consistent with the NRC staff's independent evaluations of radionuclide sorption (e.g., Leslie, et al., 2007aa, and references therein). The NRC staff notes that DOE considered uncertainty for appropriate key geochemical properties, such as pH and pCO_2, and surface properties related to sorption site concentration such as site density, surface area, and solid mass.

DOE compared the calculated surface complexation model results from almost 5,000 simulations to recent U.S. Environmental Protection Agency (EPA) compilations of soil K_ds obtained in the laboratory (SAR Section 2.3.7.12.3.4; SNL, 2007aj). DOE compared values over a pH range from 6 to 9 (SAR Section 2.3.7.12.3.4), and although the ranges are broad, there was general agreement between DOE's estimated values and the EPA compilation over the pH range considered. DOE provided reasons for potential discrepancies at pH <7, but the NRC staff observed that there was a consistent bias of calculated K_d values to the high end of the EPA ranges at pH >8. For example, for moderately sorbing radioelements such as neptunium and uranium, the calculated K_d values for $8 < pH < 9$ were about one to three orders of magnitude above the lower limit for the EPA compilation at similar pH. In DOE Enclosure 9 (2009ay), DOE provided additional experimental data that show actinides sorb more strongly onto hematite (an iron oxide) than onto clays and silicate minerals that tend to be more common in the soils assessed in the EPA compilation. In addition, analyses in DOE Enclosure 3 (2009da) show a narrower range with a lower maximum K_d value is generated in the TSPA model for all five radionuclides simulated using the surface complexation model. In effect, this TSPA approach reduces the maximum mass of actinides retained by sorption to stationary corrosion products from what might be expected from the surface complexation model results alone, increasing radionuclide concentration in the water and, all else being equal, increasing dose. Although there are differences, DOE's sorption model results are of the same order of magnitude with respect to actinide-iron oxyhydroxide sorption coefficients reported in the scientific literature. The NRC staff notes that this consistency supports the reasonableness of the K_d values used in DOE's TSPA model abstraction.

DOE used several different approaches in developing its final TSPA abstraction for sorption to stationary corrosion products and considered several different surface complexation model approaches before selecting the diffuse-layer model, as outlined in DOE Enclosure 3 (2009da). This modeling approach is widely used in the technical community, and DOE discussed its advantages and disadvantages relative to other surface complexation models. The NRC staff notes that the surface complexation model as DOE implemented it is consistent with established geochemical modeling principles, uses available experimental data to constrain chemical parameters, and is thus reasonable.

In SNL Section 6.5.2.4.2 (2007aj), DOE stated that aqueous thermodynamic data were propagated through the surface complexation model, but did not explicitly consider uncertainty in equilibrium constants for surface complexation constants used in the surface complexation model. The exclusion of this source of uncertainty is reasonable because the effects of these uncertainties on calculated radionuclide sorption would be small given the large excess number of available sorption sites as compared to aqueous radionuclide concentrations in all realizations. The large number of sorption sites in the surface complexation model for the corrosion products domain leads to the calculation of very high actinide K_d values, such that any additional uncertainty from surface complex thermodynamic data would have an insignificant effect on the transport model results. As described previously, DOE provided a reasonable evaluation of the uncertainties associated with parameters that control the number of sorption sites, such as surface area, mass of corrosion products, and site density.

DOE's abstraction assumes a single-rate, first-order kinetic model for plutonium and americium sorption. The forward rate constants for sorption of americium and plutonium onto corrosion products were estimated from plutonium sorption experiments in SNL (2008ak). Although experiments on plutonium sorption onto iron oxide colloids have been shown to be inconsistent with a single-rate model (SNL, 2007aj; Painter, et al., 2002aa), DOE provided additional information in DOE Enclosure 8 (2009ay) showing that sorption in the corrosion products domain is approximately an equilibrium process because water residence times are long compared with characteristic times for sorption. The single-rate kinetic sorption model is reasonable because the near-equilibrium condition for sorption makes details of the kinetic model irrelevant.

Colloid-Facilitated Transport

In DOE's abstraction, the transport of selected radionuclides may be enhanced by the five colloid types described in Technical Evaluation Report (TER) Section 2.2.1.3.4.3.4. DOE's mathematical models for colloid-assisted transport with reversible and irreversible sorption, with and without kinetic limitations, are consistent with approaches established in the scientific literature (e.g., Corapcioglu and Jiang, 1993aa) and the NRC staff's independent analyses (Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa). Therefore, DOE's mathematical models for colloid-assisted transport in the engineered barrier system are reasonable. Because of limited data about colloid retardation or physical straining in the engineered barrier system, the NRC staff notes that DOE appropriately excluded these retention processes when considering colloid migration in the engineered barrier system.

DOE's abstraction considers plutonium to be irreversibly associated with plutonium and zirconium colloids that originate from commercial spent nuclear fuel and montmorillonite colloids originating from defense high-level waste glass waste. Uranium is irreversibly associated with the uranophane colloids that originate in spent nuclear fuel. These three colloid types and the associated radionuclides originate in the waste form. Selected radionuclides (isotopes of americium, cesium, protactinium, plutonium, radium, tin, and thorium) are allowed to sorb reversibly and without kinetic limitations onto uranophane colloids and groundwater or waste-derived montmorillonite colloids. The empirical sorption model represents equilibrium, site-limited sorption with competition for sites. Although the assumed equilibrium distribution coefficients for plutonium are higher than the supporting data, the abstraction is reasonable because kinetically limited desorption is generally needed to significantly enhance transport for the range of colloid concentrations considered (Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa) and kinetic limitation is not modeled or expected for these colloid types. The NRC staff notes that the assumed equilibrium sorption model is consistent with current understanding of sorption on silicate minerals and is thus reasonable.

In DOE's abstraction, uranium, neptunium, and thorium sorb reversibly and without kinetic limitations to corrosion product colloids. DOE calculated equilibrium distribution coefficients (K_ds) for these elements using the same surface complexation modeling approach as for sorption on stationary corrosion products. As described in the previous subsection, the surface complexation model simulates equilibrium attachment of dissolved ions onto solid surfaces (in this case, the surfaces of mobile colloids), incorporating the chemical complexity of the system. DOE's abstraction does not include corrosion product colloids in the waste form domain. Kinetic irreversible sorption onto corrosion product colloids is modeled for americium and plutonium in the corrosion products domain. The NRC staff notes that DOE's mathematical model for colloid-associated radionuclide transport with kinetic transfers is consistent with DOE's process conceptualization, models established in the literature (e.g., Corapcioglu and Jiang, 1993aa), and the NRC staff's independent analyses (Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa).

In DOE's abstraction, the forward rate constants for americium and plutonium are not sampled directly. Instead, the fraction of total mobile radionuclide mass exiting the corrosion products domain that is associated with colloids (the target flux out ratio) is sampled from a uniform distribution with a range of 0.9 to 0.99. An analytical inverse solution is then used to calculate the forward rate corresponding to the sampled ratio. In the model abstraction, the computed forward rate constant and other parameters computed from the inverse solution are further compared against physically allowed ranges. If any computed value is outside its allowed range, the corresponding maximum or minimum value is used in the forward model for colloid-assisted transport in place of the sampled ratio. The approach based on a target flux-out ratio is reasonable because, for conditions under which colloids are stable and could potentially affect transport, this approach tends to enhance radionuclide release over that expected from a mechanistic process model.

DOE provided information suggesting that colloids will not significantly enhance transport of radionuclides in the engineered barrier system, because commercial spent nuclear fuel-derived colloids are expected to be unstable in the corrosion products domain and because radionuclide concentrations associated with iron oxide and high-level waste glass waste form colloids are expected to be smaller than dissolved concentrations. DOE's conclusion is reasonable because (i) DOE's analyses showed that commercial spent nuclear fuel-derived colloids are not expected to be stable in the pH ranges expected in the corrosion products domain (DOE, 2009ay); (ii) the NRC staff's confirmatory calculations (TER Section 2.2.1.3.4.3.4) show that iron oxide colloids exert negligible control on dissolved plutonium concentrations; and (iii) the mean sampled plutonium concentration associated with high-level waste glass colloids is an order of magnitude lower than the mean plutonium solubility limit (TER Section 2.2.1.3.4.3.4).

Reasonableness and Consistency of TSPA Results for Engineered Barrier System Radionuclide Releases

The NRC staff performed simplified confirmatory analyses to assess whether DOE's Total System Performance Assessment (TSPA) results for the engineered barrier system radionuclide releases are consistent with DOE's abstractions. As discussed in the following paragraphs, the NRC staff used hand calculations (Painter, 2010aa; Pickett, 2010aa) to estimate peak expected total-repository engineered barrier system release rates for Tc-99, Pu-242, and Np-237 using DOE-provided information, and then compared these releases with DOE-provided values.[1] The NRC staff's confirmatory calculations focused on the igneous intrusion and seismic ground motion modeling cases because these cases result in the largest release rates from the engineered barrier system. These estimates are based on advection and diffusion of dissolved radionuclides and neglect transport of radionuclides associated with colloids. As discussed in TER Section 2.2.1.3.4.3.4, transport in the engineered barrier system is not significantly enhanced by colloids, because of low colloid concentrations in the corrosion products domain. The NRC staff's simplified calculation estimates are presented only for selected cases for which representative DOE results were available for comparison. Results are presented for waste package release rates (i) of Pu-242 and Np-237 in the igneous intrusion modeling case, on the basis of control of the dissolved concentration limited by solubility (TER Section 2.2.1.3.4.3.3) or corrosion product sorption (this TER Section 2.2.1.3.4.3.5), and (ii) of Tc-99 in the seismic ground motion modeling case, on the basis of release rate control by waste package failure rate.

DOE provided information showing that engineered barrier system releases of low-solubility, sorbing radionuclides (e.g., plutonium and neptunium) are mainly controlled by processes within the corrosion products domain because waste form dissolution and invert transport processes are fast relative to transport within the corrosion products domain. In the TSPA analyses, the important dose contributions from plutonium and neptunium isotopes result from the igneous intrusion modeling case (SAR Section 2.4.2.2.1.1.3), in which all waste packages fail and releases of these radionuclides are controlled by advection modified by sorption and precipitation of radionuclide-bearing minerals. The NRC staff performed simplified estimates to confirm DOE's release calculations for Pu-242 and Np-237 for the igneous intrusion

[1]The NRC staff's calculations are estimates only, using parameters and probabilities DOE provided in the SAR, supporting reports, and associated responses to requests for additional information. Because simplifications are involved, precise agreement between the NRC staff calculations and DOE's results was not expected; rather, reasonable agreement of the release rates within an order of magnitude was evaluated.

modeling case in the engineered barrier system. The NRC staff assumed that these release rates for the corrosion products domain are controlled by advection and either (i) precipitation of solubility-limiting minerals or (ii) sorption onto corrosion products. The solubility limit (TER Section 2.2.1.3.4.3.3) is a chemically based maximum value for the dissolved concentration of an element, in the absence of sorption. If, however, there is capacity for sorbing the dissolved element onto solid surfaces, the dissolved concentration may not reach the solubility limit and could be controlled to a lower value by sorption. Because both solubility and sorption are viable processes in the corrosion products domain, the process that limits the dissolved concentration to the lowest value would provide the better simplified estimate of release rate. The NRC staff, therefore, used simplified calculations to estimate release rates controlled by both potential limits on dissolved concentration—solubility and sorption—and selected the lower value for comparison to DOE's results.

Using information provided by DOE, the NRC staff estimated a peak mean, repositorywide, sorption-limited Pu-242 engineered barrier system release rate of 7.9 g/yr [0.017 lb/yr] (Painter, 2010aa; Pickett, 2010aa). The NRC staff calculated a solubility-limited Pu-242 release rate of 1.2 g/yr [0.0026 lb/yr] (Painter, 2010aa; Pickett, 2010aa), which is lower than the sorption-limited rate and is, therefore, taken as the estimated value.

DOE provided representative commercial spent nuclear fuel package release results for percolation subarea 3 (which includes approximately 40 percent of the waste packages) and a single realization from the igneous intrusion case in DOE Enclosure 1, Figure 2 (2009da). For Pu-242, the peak corrosion products domain release from subarea 3 is approximately 30 g/yr [0.07 lb/yr] at 100,000 years, conditional on an igneous event having occurred at 10,000 years. Scaling that value to the full repository and multiplying by the probability of having a single igneous intrusion event in 1 million years (1.7 percent) gives a value of 1.3 g/yr [0.0029 lb/yr]. This DOE result is consistent with the NRC staff's confirmatory estimate of 1.2 g/yr [0.0026 lb/yr].

The NRC staff calculated an estimated peak mean, repositorywide, sorption-limited release rate for Np-237 of 3.0 g/yr [0.0066 lb/yr] for the igneous intrusion modeling case (Painter, 2010aa; Pickett, 2010aa). DOE provided representative results for Np-237 release from the corrosion products domain in percolation subarea 3 and a single realization of the igneous intrusion modeling case in DOE Figure 1.1-12 (2009dc). That realization assumes that an igneous intrusion event occurs. Thus, DOE's calculated peak release rate of approximately 200 g/yr [0.4 lb/yr] is weighted by the probability of an igneous event in 1 million years (1.7 percent) and scaled to the full repository to compare with the NRC staff's peak mean estimate. The result of that calculation is 8.5 g/yr [0.019 lb/yr] for Np-237 release for the igneous intrusion modeling case. This DOE result is in reasonable agreement with the NRC staff's simplified calculation estimate of 3.0 g/yr [0.0066 lb/yr].

The NRC staff calculated an NpO_2 solubility-limited neptunium engineered barrier system release rate of 12 g/yr [0.026 lb/yr] (Painter, 2010aa, Pickett, 2010aa). This result is higher than the sorption-limited rate and was therefore not used for comparison; as discussed previously in this section, the lower calculated rate is more appropriate for comparisons. A rate calculated using the Np_2O_5 solubility model (applicable after all in-package steel has corroded) would be even higher and was therefore not used for comparison.

For high-solubility radionuclides that are weakly sorbing or nonsorbing (e.g., technetium), radionuclide-specific engineered barrier system releases in the seismic ground motion and nominal modeling cases are controlled primarily by the waste package failure rate because dissolution rates of the waste form and transport of these radionuclides through the engineered barrier system are sufficiently fast compared with typical intervals between package failures (TER Section 2.2.1.4.1.3.3.1.2). Using the information provided by DOE, the NRC staff calculated a peak mean, repositorywide release rate for Tc-99 of 22 g/yr [0.049 lb/yr] for the seismic ground motion modeling case (Painter, 2010aa; Pickett, 2010aa). DOE provided plots of mean cumulative Tc-99 engineered barrier system release in SAR Figure 2.1-24. From the slopes of these plots, the NRC staff estimated that the release rate peaks at approximately 12 g/yr [0.026 lb/yr] for the period between 10,000 and 100,000 years. This peak mean rate estimated from DOE's Tc-99 release information is in reasonable agreement with the NRC staff's simplified calculated estimate of 22 g/yr [0.049 lb/yr] (Painter, 2010aa; Pickett, 2010aa).

Summary of NRC Staff's Review of Engineered Barrier System Radionuclide Transport

The NRC staff notes that, in modeling the transport of radionuclides in the engineered barrier system, DOE adequately described the system and models used, applied appropriate conceptual models, and considered alternative conceptual models. DOE used appropriate mathematical models to represent transport in the engineered barrier system. Transfer of information between the radionuclide transport abstraction and other TSPA code abstractions was consistently implemented. Relevant design information for the waste package was incorporated. DOE used appropriate data to establish model parameters and to represent uncertainty. Intermediate results of the abstraction were compared to independent information.

Release rates DOE's TSPA model produced for radionuclide transport in the engineered barrier system vary significantly by radionuclide and modeling case. The engineered barrier system does not significantly delay transport of soluble, nonsorbing radionuclides such as Tc-99 and I-129, and the waste package failure rates control the engineered barrier system release rates for those radionuclides. Transport of low-solubility, sorbing radionuclides such as Np-237 and Pu-242 is significantly slower and is generally controlled by sorption onto stationary corrosion products and precipitation of radionuclide-bearing minerals in the corrosion products domain. Colloid-assisted transport is not significant compared with transport of dissolved radionuclides, because of limited colloid concentrations in the engineered barrier system. TSPA code results for engineered barrier system release rates DOE provided are consistent with the NRC staff's simplified confirmatory calculations. Because the staff's simplified confirmatory calculations are consistent with DOE's TSPA code results, the NRC staff notes that the TSPA results are reasonable.

2.2.1.3.4.4 NRC Staff Conclusions

The NRC staff notes that the DOE description of this model abstraction for radionuclide release rates and solubility limits is consistent with the guidance in the YMRP. The NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.4.5 References

Allison, J.M. 2004aa. "Request for Referenceable Information on High-Level Waste (HLW) Radionuclide Inventories in Support of Preparation of the Yucca Mountain Project License Application, JCP–0445, 1/28/04." Memorandum (February 26, 0303040661, with attachment) to J. Authur, III, OCRWM. MOL20040317.0265. Aiken, South Carolina: DOE, Savannah River Operations Office.

Appelo, C.A.J. and Postma, D. 1994aa. *Geochemistry, Groundwater and Pollution*. Brookfield, Vermont: A.A. Balkema.

Beavers, J.A. and C.L. Durr. 1990aa. *Container Corrosion in High Level Nuclear Waste Repositories*. Columbus, Ohio: Cortest Columbus, Inc.

Bowman, S.M. and L.C. Leal. 2000aa. NUREG/CR–0200, "ORIGEN-ARP: Automatic Rapid Process for Spent Fuel Depletion, Decay, and Source Term Analysis." ORNL/NUREG/CSD–2/V1/R6. Oak Ridge, Tennessee: Oak Ridge National Laboratory.

Bremier, S., C.T. Walker, and R. Menzel. 2000aa. "Fission Gas Release and Fuel Swelling at Burn-Ups Higher Than 50 Mwd/kgU." Proceedings of the Fission Gas Behavior in Water Reactor Fuels. Cadarache, France, September 26–29, 2000. Paris, France: Nuclear Energy Institute. pp. 93–106.

BSC. 2005ad. "In-Package Chemistry Abstraction." ANL–EBS–MD–000037. Rev. 04. ACN 01, AD 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ae. "Aqueous Corrosion Rates for Waste Package Materials." ANL–DSD–MD–000001. Rev. 01. ACN 001, ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ah. "CSNF Waste Form Degradation: Summary Abstraction." ANL–EBS–MD–000015. Rev. 02. ACN 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ai. "Defense HLW Glass Degradation Model." ANL–EBS–MD–000016. Rev. 02. ACN 001, ERD 001, ERD 002. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ao. "DSNF and Other Waste Form Degradation Abstraction." ANL–WIS–MD–000004. Rev. 4. Las Vegas, Nevada: Bechtel SAIC Company, LLC.
BSC. 2003af. "PWR Assembly End-Effect Reactivity Evaluation." CAL–UDC–NU–000006. Rev. 00, with 01 errata. MOL20010412.0158. DOC20031014.0007. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Corapcioglu, M.Y. and S. Jiang. 1993aa. "Colloid-Facilitated Groundwater Contaminant Transport." *Water Resources Research*. Vol. 29, No. 7. pp. 2,215–2,226.

Cvetkovic, V., S. Painter, D. Turner, D. Pickett, and P. Bertetti. 2004aa. "Parameter and Model Sensitivities for Colloid-Facilitated Transport on the Field Scale." *Water Resources Research*. Vol. 40. doi:10.1029/2004WR003048.

Da Cunha Belo, M., M. Walls, N.E. Hakiki, J. Corset, E. Picquenard, G. Sagon, and D. Noel. 1998aa. "Composition, Structure, and Properties of Oxide Films Formed on the Stainless Steel 316L in a Primary Type PWR Environment." *Corrosion Science*. Vol. 40, Nos. 2 and 3. pp. 447–463.

DOE. 2010aj. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 1." Letter (February 24) J.R. Williams to J.H. Sulima (NRC). ML100560258. Washington, DC: DOE, Office of Technical Management.

DOE. 2009an. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1.” Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Washington, DC: DOE, Office of Technical Management.

DOE. 2009av. DOE/RW–0573, “Yucca Mountain Repository License Application.” Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009ax. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report, Vol. 3, Chapter 2.2.1.3.4, Set 1.” Letter (May 5) J.R. Williams to J.H. Sulima (NRC). ML091260473. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ay. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report, Vol. 3, Chapter 2.2.1.3.4, Set 2.” Letter (May 12) J.R. Williams to J.H. Sulima (NRC). ML 091330282. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cz. “Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 1 and Set 2.” Letter (June 26) J.R. Williams to J.H. Sulima (NRC). ML091770582. Washington, DC: DOE, Office of Technical Management.

DOE. 2009da. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 3.” Letter (September 11) J.R. Williams to J.H. Sulima (NRC). ML092600883. Washington, DC: DOE, Office of Technical Management.

DOE. 2009db. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 4.” Letter (October 16) J.R. Williams to J.H. Sulima (NRC). ML093200320. Washington, DC: DOE, Office of Technical Management.

DOE. 2009dc. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 2.” Letter (May 28) J.R. Williams to J.H. Sulima (NRC). ML091480752. Washington, DC: DOE, Office of Technical Management.

DOE. 2009dd. “Yucca Mountain—Supplement Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.7), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.4, Set 2.” Letter (September 4) J.R. Williams to J.H. Sulima (NRC). ML092470427. Washington, DC: DOE, Office of Technical Management.

DOE. 2003ad. “Review of DOE Spent Nuclear Fuel Release Rate Test Results.” Rev. 0. DOE/SNF/REP–073. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory.

DOE. 2000aa. DOE/SNF/REP–056. “N Reactor (U-Metal) Fuel Characteristics for Disposal Criticality Analysis.” Rev. 0. TIC247956. Washington, DC: DOE, Office of Environmental Management.

Finch, R.J., E.C. Buck, P.A. Finn, and J.K. Bates. 1999aa. “Oxidative Corrosion of Spent UO2 Fuel in Vapor and Dripping Groundwater at 90 °C.” 22nd Symposium on Scientific Basis for Nuclear Waste Management, 1998 Material Research Society Fall Meeting, Boston, Massachusetts, November 30–December 4, 1998. Symposium Proceedings Vol. 556. Warrendale, Pennsylvania: Materials Research Society. p. 431.

Glass, R.S., G.E. Overturf, R.E. Garrison, and R.D. McCright. 1984aa. “Electrochemical Determination of the Corrosion Behavior of Candidate Alloys Proposed for Containment of High Level Nuclear Waste in Tuff.” Report UCID–20174. Livermore, California: Lawrence Livermore National Laboratory.

GoldSim Technology Group. 2006aa. *GoldSim Contaminant Transport Module User’s Guide. Version 4.0*. Issaquah, Washington: GoldSim Technology Group.

Gray, W.J. 1999aa. “Inventories of Iodine-129 and Cs-137 in the Gaps and Grain Boundaries of LWR Spent Fuels.” Proceedings of the Materials Research Society. Symposium Proceedings 556. Warrendale, Pennsylvania: Materials Research Society. pp. 487–494.

Harrar, J.E., J.F. Carley, W.F. Underwood, and E. Raber. 1990aa. “Report of the Committee to Review the Use of J-13 Well Water in Nevada Nuclear Waste Storage Investigations.” UCID–21867. ACC:NNA 19910131.0274. Livermore, California: Lawrence Livermore National Laboratory.

Hem, J.D. 1995aa. *Study and Interpretation of the Chemical Characteristics of Natural Water*. 3rd Edition. Geological Survey Water-Supply Paper 2254. Washington, DC: U.S. Government Printing Office.

Jain, V., G. Cragnolino, and L. Howard. 2004aa. “A Review Report on High Burnup Spent Nuclear Fuel—Disposal Issues.” ML 042860337. San Antonio, Texas: CNWRA.

Jantzen, C.M., D.I. Kaplan, N.E. Bibler, D.K. Peeler, and M.J. Plodinec. 2008aa. “Performance of a Buried Radioactive High-Level Waste (HLW) Glass after 24 Years.” *Journal of Nuclear Materials*. Vol. 378. pp. 244–256.

Johnson, L.H. and J.C. Tait. 1997aa. “Release of Segregated Nuclides From Spent Fuel.” SKB Technical Report 97-18. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Company.

Langmuir, D. 1997aa. *Aqueous Environmental Geochemistry*. Upper Saddle River, New Jersey: Prentice Hall.

Lassmann, K., C.T. Walker, J. ven de Larr, and F. Lindstrom. 1995aa. “Modeling the High Burnup UO2 Structure in LWR Fuel.” *Journal of Nuclear Materials*. Vol. 226. pp. 1–8.

Leslie, B., C. Grossman, and J. Durham. 2007aa. “Total-system Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide.” Rev. 1. ML072710060. San Antonio, Texas: CNWRA.

MacKinnon, R.J. 2008aa. "Engineered Barrier System (EBS) Transport Model Abstractions." Presentation at DOE/NRC Technical Exchange on Total System Performance Assessment (TSPA) for Yucca Mountain, Las Vegas, Nevada, April 4, 2008. LSN:DEN001594514. Washington, DC: NRC.

Manaktala, H. 1993aa. "Characteristics of Spent Nuclear Fuel and Cladding Relevant to High-Level Waste Source Term." CNWRA 93-006. ML 040200340. San Antonio, Texas: CNWRA.

McCright, R.D., W.G. Halsey, and R.A. Van Konynenburg. 1987aa. "Progress Report on the Results of Testing Advanced Conceptual Design Metal Barrier Materials Under Relevant Environmental Conditions for a Tuff Repository." Report No. UCID–21044. Livermore, California: Lawrence Livermore National Laboratory.

Mohanty, S., G. Adams, and R. Pabalan. 2003aa. "The Role of Solubility as a Barrier to Radionuclide Release." Proceedings of the 10th International High-Level Radioactive Waste Management Conference, Las Vegas, Nevada, March 30–April 2, 2003. La Grange Park, Illinois: American Nuclear Society. pp. 938–945.

Morgenstein, M.D., C.L. Wicket, and A. Barkatt. 1999aa. "Consideration of Hydration-rind Dating of Glass Artefacts: Alteration Morophologies and Experimental Evidence of Hydrogeochemical Soil-Zone Pore Water Control." *Journal of Archaeological Science*. Vol. 26. pp. 1,193–1,210.

Murphy, W.M. and R.C. Codell. 1999aa. "Alternate Source Term Models for Yucca Mountain Performance Assessment Based on Natural Analog Data and Secondary Mineral Solubility." Proceedings of the Materials Research Society, November 30–December 4, 1998, Boston, Massachusetts. Symposium Proceedings 556. D.J. Wronkiewicz and J.H. Lee, eds. Warrendale, Pennsylvania: Materials Research Society. pp. 551–558.

Neck, V., M. Altmaier, A. Seibert, J.I. Yun, C.M. Marquardt, and T. Fanghänel. 2007aa. "Solubility and Redox Reactions of Pu(IV) Hydrous Oxide: Evidence for the Formation of PuO_{2+x}(s, hyd)." *Radiochimica Acta*. Vol. 95. pp. 193–207.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2008aa. NUREG–1914, "Dissolution Kinetics of Commercial Spent Nuclear Fuels in the Potential Yucca Mountain Repository Environment." ML 083120074. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC. 1996ab. NUREG–1564, "Long-Term Kinetic Effects and Colloid Formation in the Dissolution of LWR Spent-Fuel." ML 073100056. Washington, DC: NRC.

Nuclear Energy Agency. 1997aa. "Lessons Learnt from Ten Performance Assessment Studies." Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Painter, S. 2010aa. "Radionuclide Transport Analysis for Yucca Mountain." Electronic Scientific Notebook 318E. San Antonio, Texas: CNWRA.

Painter, S. and V. Cvetkovic. 2006aa. "Effect of Kinetic Limitations on Colloid-Facilitated Transport at the Field Scale." 11th International High-Level Radioactive Waste Management Conference, Global Progress Toward Safe Disposal, Las Vegas, Nevada, April 30–May 4, 2006. LaGrange Park, Illinois: American Nuclear Society.

Painter, S., V. Cvetkovic, D. Turner, and D. Pickett. 2002aa. "Significance of Kinetics for Sorption on Inorganic Colloids: Modeling and Experimental Interpretation Issues." *Environmental Science and Technology*. Vol. 24. pp. 228–234.

Pearcy, E.C., J.D. Prikryl, W.M. Murphy, and B.W. Leslie. 1994aa. "Alteration of Uraninite From the Nopal I Deposit, Peña Blanca District, Chihuahua, Mexico, Compared to Degradation of Spent Nuclear Fuel in the Proposed U.S. High-Level Nuclear Waste Repository at Yucca Mountain, Nevada." *Applied Geochemistry*. Vol. 9. pp. 713–732.

Pickett, D.A. 2010aa. "Confirmatory Calculations of Radionuclide Release Rates and Colloid Transport Supporting Yucca Mountain Review." Electronic Scientific Notebook 1052E. San Antonio, Texas: CNWRA.

Pickett, D.A. 2005aa. "Effects of Spent Nuclear Fuel Uranyl Alteration Phases on Radionuclide Dissolved Concentration Limits." San Antonio, Texas: CNWRA.

Prikryl, J.D. 2008aa. "Uranophane Dissolution and Growth in $CaCl_2$–SiO_2(aq) Test Solutions." *Geochimica et Cosmochimica Acta*. Vol. 72. pp. 4,508–4,520.

Pulvirenti, A.L., S.J. Eddy, T.M. Calabrese, M.A. Adel-Hadadi, A. Barkatt, and M.E. Morgenstein. 2006aa. "Interaction of Iron Containing Silicate Glasses with Aqueous Salt Solutions." *Physics and Chemistry of Glasses: European Journal of Glass Science and Technology Part B*. Vol. 47, No. 1. pp. 47–57.

Rai, D., N.J. Hess, A.R. Felmy, D.A. Moore, M. Yui, and P. Vitorge. 1999aa. "A Thermodynamic Model for the Solubility of PuO_2(am) in the Aqueous K^+-HCO_3^--CO_3^{2-}-OH^--H_2O System." *Radiochimica Acta*. Vol. 86. pp. 89–99.

Roxburgh, I.S. 1987aa. *Geology of High-Level Nuclear Waste Disposal: An Introduction*. New York City, New York: Chapman and Hall.

Shoesmith, D.W. 2000aa. "Fuel Corrosion Processes under Waste Disposal Conditions." *Journal of Nuclear Materials*. Vol. 282. pp. 1–31.

SNL. 2008ab. "Features, Events, and Processes for the Total System Performance Assessment: Analyses." ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ad. "Postclosure Nuclear Safety Design Bases." ANL–WIS–MD–000024. Rev. 01. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ak. "Waste Form and In-Drift Colloids-Associated Radionuclide Concentrations: Abstraction and Summary." MDL–EBS–PA–000004. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ag. "Dike/Drift Interactions." MDL–MGR–GS–000005. Rev. 02. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ah. "Dissolved Concentration Limits of Elements with Radioactive Isotopes." ANL-WIS–MD–000010. Rev. 06. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aj. "EBS Radionuclide Transport Abstraction." ANL–WIS–PA–000001. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ak. "Engineered Barrier System: Physical and Chemical Environment." ANL–EBS–MD–000033. Rev. 06. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ao. "In-Drift Precipitates/Salts Model." ANL–EBS–MD–000045. Rev. 03. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007at. "Qualification of Thermodynamic Data for Geochemical Modeling of Mineral—Water Interactions in Dilute Systems." ANL–WIS–GS–000003. Rev. 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bm. "Total System Performance Assessment Data Input Package for Requirements Analysis for DOE SNF/HLW and Naval SNF Waste Package Overpack Physical Attributes Basis for performance Assessment." Rev. 00. TDR–TDIP–ES–000003. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bn. "Total System Performance Assessment Data Input Package for Requirements Analysis for TAD Canister and Related Waste Package Overpack Physical Attributes Bases for Performance Assessment." Rev. 00. TDR–TDIP–ES–000006. Las Vegas, Nevada: Sandia National Laboratories.

Spino, J., J. Cobos-Sabate, and F. Rousseau. 2003aa. "Room-Temperature Microindentation Behaviour of LWR-Fuels, Part 1: Microhardness." *Journal of Nuclear Materials*. Vol. 322. pp. 204–216.

van de Weerd, H., A. Leijnse, and W.H. van Riemsdijk. 1998aa. "Transport of Reactive Colloids and Contaminants in Groundwater: Effect of Nonlinear Kinetic Interactions." *Journal of Contaminated Hydrology.* Vol. 32. pp. 313–331.

Wang, X.Y., Y.S. Wu, L. Zhang, and Z.Y. Yu. 2001aa. "Atomic Force Microscopy and X-Ray Photoelectron Spectroscopy Study on the Passive Film for Type 316L Stainless Steel." *Corrosion Science*. Vol. 57, No. 6. pp. 540–546.

Wilson, C.N. and W.J. Gray. 1990aa. "Measurement of Soluble Nuclide Dissolution Rates From Spent Fuel." Proceedings of the Materials Research Society. Symposium Proceedings 176. V.M. Oversby and P.W. Brown, eds. Warrendale, Pennsylvania: Materials Research Society. pp. 489–498.

Wronkiewicz, D.J., K.J. Bates, S.F. Wolf, and E.C. Buck. 1996aa. "Ten-Year Results From Unsaturated Drip Tests With UO_2 at 90°C: Implications for the Corrosion of Spent Nuclear Fuel." *Journal of Nuclear Materials*. Vol. 238, No. 1. pp. 78–95.

Zhu, C. 2004aa. "Coprecipitation in the Barite Isostructural Family: 2. Numerical Simulations of Reactions and Mass Transport." *Geochimica et Cosmochimica Acta*. Vol. 68. pp. 3,339–3,349.

CHAPTER 8

2.2.1.3.5 Climate and Infiltration

2.2.1.3.5.1 Introduction

This chapter of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE) representation of climate and infiltration, as presented in DOE's Safety Analysis Report (SAR) Section 2.3.1 (DOE, 2008ab) and supporting information. DOE considers the reduction of water flux from precipitation to net infiltration to be a barrier capability for the Upper Natural Barrier. Because of the generally vertical movement of percolating water through the unsaturated zone in the DOE representation of the natural system, water entering the unsaturated zone at the ground surface (infiltration) is the only source for deep percolation water in the unsaturated zone at and below the proposed repository.

DOE used the term "net infiltration" to define the volumetric flux of water passing below the active plant root zone, but often refers to net infiltration simply as "infiltration." DOE also used the term net infiltration to refer both to the output of the net infiltration model (SAR Section 2.3.1) and to the top boundary condition of the unsaturated zone model (SAR Section 2.3.2). This distinction is important because the average values from the net infiltration model differ from those used as net infiltration at the top boundary of the site-scale unsaturated zone model. NRC staff evaluates the former in the present section and the latter in TER Section 2.2.1.3.6.3.2.

Climate and infiltration are treated differently in DOE's performance assessment for the first 10,000 years following permanent closure of the repository and the period from 10,000 to 1 million years. For the first 10,000 years, DOE used paleoclimate records for the region to predict future climatic conditions and uses these predictions as input for estimating future net infiltration. In addition, DOE used the climate predictions to scale groundwater fluxes in the saturated zone portion of the performance assessment for this period. DOE described its approach for scaling the groundwater flux in SAR Section 2.3.9; the NRC staff's review of groundwater flux in the saturated zone is in TER Section 2.2.1.3.8. For the period from 10,000 years to 1 million years after permanent closure, 10 CFR 63.342(c)(2) allows DOE to consider long-term-average deep percolation flux at the proposed repository horizon instead of explicitly predicting climate and infiltration. DOE chose to use the prescribed deep percolation flux in its performance assessment for the post-10,000-year period. In TER Section 2.2.1.3.6.3.2, NRC staff evaluates DOE's use of the prescribed average deep percolation flux at the proposed repository horizon for post-10,000-year performance assessment calculations.

This TER chapter provides the NRC staff's evaluation of DOE's consideration of climate and infiltration in the first 10,000 years after closure in DOE's Total System Performance Assessment calculations. NRC staff reviewed the DOE technical bases, input data, models, and net infiltration results. The NRC staff used its understanding of relative risk within the repository system to inform its review, by focusing on those aspects that are most significant for repository performance. The NRC staff considered how the flux of water through the unsaturated zone affects seepage, release of radionuclides from the engineered barrier system, and radionuclide transport through the natural system (evaluated in TER Sections 2.2.1.3.6.3.4, 2.2.1.3.4.3.5, and 2.2.1.3.7, respectively) in determining the significant aspects of DOE's net infiltration model. On the basis of the downstream uses of climate and infiltration calculations, staff focused its review on DOE's estimates of the magnitude, spatial distribution, and uncertainty of net infiltration over the next 10,000 years.

2.2.1.3.5.2 Evaluation Criteria

NRC staff's review of model abstractions used in the DOE postclosure performance assessment, including those considered in this chapter for climate and infiltration, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction includes changes in future climate and the effect of such changes on infiltration of water. 10 CFR 63.342(c)(2) provides requirements for assessing the effects of climate change. DOE has chosen to assess the effects of climate change for the period after 10,000 years following disposal by using constant-in-time deep percolation rates in its performance assessment. The NRC staff's evaluation of DOE's use of these deep percolation rates is in TER Section 2.2.1.3.6.3.2.

In addition,10 CFR 63.305 provides the following requirements for characteristics of the reference biosphere as used in this abstraction for climate and infiltration:

- Features, events, and processes that describe the reference biosphere are consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site [10 CFR 63.305(a)].

- DOE should not project changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology; all analyses should assume that all of those factors are constant as they are at the present time [10 CFR 63.305(b)].

- Factors related to the geology, hydrology, and climate should vary based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, consistent with the requirements for performance assessments specified at 10 CFR 63.342 [10 CFR 63.305(c)].

- Biosphere pathways are consistent with arid or semi-arid conditions [10 CFR 63.305(d)].

The requirements of 10 CFR 63.305 apply to the abstractions for climate and infiltration to the extent that the characteristics of the reference biosphere affect climate and infiltration.

The NRC staff's review of the SAR follows the guidance laid out in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Section 2.2.1.3.5, Climate and Infiltration, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of climate and infiltration are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff review used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of climate and infiltration important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE, and on NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.5.3 Technical Evaluation

The review of the technical information DOE provided for climate and infiltration over the next 10,000 years is divided into three subsections within this TER section. The first subsection reviews DOE's identification and description of features and processes for climate and infiltration. The second subsection focuses on the climate data, future-climate model, and climate predictions. The third subsection addresses DOE's description of net infiltration processes, models, and estimates of net infiltration at Yucca Mountain using the future climate predictions.

2.2.1.3.5.3.1 Identification of Features and Processes

In this section, NRC staff evaluates the DOE identification and description of processes important for estimating climate and net infiltration. This section addresses the system description and model integration. DOE's overall screening of FEPs is reviewed in TER Section 2.2.1.2.1.

DOE used regional and site characteristics to develop conceptual models for climate and net infiltration at Yucca Mountain. The natural features of topography and surficial soils of the Upper Natural Barrier were identified as important to waste isolation in SAR Section 2.1.2.1. On the basis of field observations, synthesis of data, and modeling over more than two decades, DOE indicated (SAR Section 2.3.1.1) that the features and processes important to the capability of the Upper Natural Barrier are (i) climate change, (ii) climate modification increases recharge, (iii) precipitation, (iv) topography and morphology, (v) rock properties of host rock and other units, (vi) surface runoff and evapotranspiration, (vii) infiltration and recharge, (viii) fractures, and (ix) fracture flow in the unsaturated zone.

The following summary is based on the information in SAR Section 2.3.1.1 and illustrates how DOE integrated these features and processes in its conceptual models of climate and infiltration. DOE described the present climate at Yucca Mountain as semi-arid, with low annual precipitation. DOE expects the climate to change over the next 10,000 years, remaining semi-arid but with changes in precipitation patterns and rates. DOE recognized that surface temperature and vegetation will also vary with changes in climate. Evapotranspiration (the combination of evaporation and plant transpiration) removes a large portion of the annual precipitation that infiltrates into the soil. In this environment, evapotranspiration is strongly influenced by temperature and low atmospheric relative humidity. DOE conceptualized that net infiltration events occur in pulses during and for a short period following some of the larger or longer duration precipitation events. Evapotranspiration continually dries the soil between precipitation events. DOE considered snow as providing a source of delayed infiltration during snowmelt events. In DOE's conceptual model, runoff and the soil's water-holding capacity limit the magnitude of net infiltration pulses, but runoff from one area may subsequently infiltrate downstream.

DOE conceived that soil, fractures, and bulk rock hydraulic properties affect the rate of water movement below the root zone, with a competition between downward flow and upward movement of water by evapotranspiration. In the DOE conceptual model, water flows quickly into the rock below shallow soil in areas where the bedrock is fractured or has a highly permeable matrix. Such rapid flow limits the effect of evapotranspiration. Surface water runoff, influenced by topography and surface morphology, spatially redistributes the flux of water. This process may reduce net infiltration in some areas (e.g., high on hillslopes) and increase net infiltration in others (e.g., washes and channels). DOE recognized that lateral movement of water below the surface, termed interflow, is known to spatially redistribute water, but the DOE conceptual model does not consider interflow to be significant at Yucca Mountain. For the semi-arid climate of Yucca Mountain, the overall water balance in DOE's model is dominated by precipitation and evapotranspiration, with infiltration and runoff representing relatively small portions of the balance.

DOE implemented its conceptualization in (i) a climate model that predicts future climatic states, (ii) climatic input data for each climate state, and (iii) an infiltration model linked to a surface-water-routing algorithm for runoff. The infiltration and routing algorithms are integrated into the Mass Accounting System for Soil Infiltration and Flow (MASSIF) model. DOE described

- A climate model for predicting climate over the next 10,000 years that uses Earth-orbital parameters and paleoclimatic data from the southwestern United States covering the past approximately 800,000 years

- Climatic input data for each climate state that uses recorded meteorological data from local, regional, and western U.S. stations

- Submodels of the infiltration model that consider precipitation, evapotranspiration, snowmelt, runoff and run on, and infiltration

DOE used site characterization data, as available, to develop inputs for the MASSIF model (SAR Section 2.3.1.3). Where sparse or no site observations are available, other information from scientific literature or other sites was used to develop inputs.

DOE reasonably identified and included features and processes in its climate models that are important for estimating future climatic conditions at Yucca Mountain. This is supported by a comparison of the information provided by DOE with staff's knowledge of past climates in the southwest United States, including the Yucca Mountain region, obtained from literature reviews and independent analysis (NRC, 2005aa; Stothoff and Musgrove, 2006aa; Stothoff and Walter, 2007aa). NRC staff compared DOE's description of infiltration, and the incorporation of features and processes, with NRC staff's understanding of near-surface features and processes at Yucca Mountain obtained from literature reviews, field observations, and independent analysis (NRC, 2005aa; Stothoff, 2008aa,ab, 2009aa). Because DOE's description of infiltration and incorporation of features and processes are consistent with those from other sites and with staff's independent analyses (see previous), DOE reasonably identified and included in its overall conceptual model features and processes important for net infiltration at Yucca Mountain.

2.2.1.3.5.3.2 Climate

This section contains NRC staff's review of the DOE approach and results for predicting climate states for the next 10,000 years, and for predicting climatic conditions within each of the climate states. The NRC staff evaluated the performance assessment calculations representing the first 10,000 years of repository performance, to assess DOE's net infiltration estimates. In its performance-based review, NRC staff focused on identifying whether the data, models, and results represent climate and the uncertainty of predicting future climate conditions. Because DOE chose to use the range and distribution of average deep percolation specified in 10 CFR Part 63 for the period from 10,000 to 1 million years after closure, DOE did not provide information on climate or meteorology during this period. Therefore, the NRC staff did not evaluate explicit models for climate or meteorology for the post-10,000-year period in its TER.

The NRC staff evaluates the DOE approach to representing deep percolation during the period from 10,000 to 1 million years after closure in TER Section 2.2.1.3.6.3.2.

The NRC staff evaluates DOE's approach to modeling climate during the first 10,000 years by separately considering DOE's approach to estimating long-term-average climate (TER Section 2.2.1.3.5.3.2.1) and DOE's approach to estimating daily weather parameters given a long-term-average climate state (TER Section 2.2.1.3.5.3.2.2). These two sections separate climatic considerations into long and short time scales, respectively.

2.2.1.3.5.3.2.1 Climate Change for the Next 10,000 Years

This section addresses data for model justification, and characterization and propagation of data uncertainty, for DOE's climate prediction for the 10,000 years following repository closure.

DOE predicted climate states covering the next 10,000 years using paleoclimate proxies from regional records and the understanding of orbital variations as the principal drivers of Earth climate over the past several million years. DOE's main paleoclimate proxies are layered mineral precipitates from Devils Hole and fossils preserved in continuously layered lake sediments from Owens Lake. Owens Lake, a present-day playa, and Devils Hole, a water-filled cave, are both within 140 km [87 mi] of Yucca Mountain. Cores from both sites record past cyclic changes in regional climate, between glacial and interglacial phases, and are generally consistent with other global climate proxy records.

From these records, DOE derived three representative states for future climate. DOE predicted these three climates and the timing of step changes by (i) identifying the past point in time in the Devils Hole record that is equivalent to the present moment within the glacial cycle, (ii) identifying the same equivalent point in the Owens Lake sediment sequence, (iii) identifying the sediment sequence corresponding to the 10,000 years following the equivalent point, and finally, (iv) attributing climate states to the sediment sequence (SAR Section 2.3.1.2.3.1.1).

The climate-sequence timing DOE described is based on two Earth-orbital parameters, which are recognized as climate forcing functions operating over geologic time scales: orbital eccentricity, with a period of approximately 100,000 years, and precession of the equinoxes, with a period of approximately 23,000 years (SAR Section 2.3.1.2.1.2.3). Values for these orbital parameters can be calculated to high precision from astronomical relations. DOE used oxygen isotope ratio ($\delta^{18}O$) records in Devils Hole vein calcite, dated using uranium-series methods, to relate the orbital parameters to past glacial stages. In the SAR, DOE explained that obliquity, another recognized orbital forcing with a period of approximately 41,000 years, was not used in its model, because no consistent relationship was shown between obliquity and the Devils Hole $\delta^{18}O$ record. The SAR asserted that groups of 4 eccentricity cycles, totaling approximately 400,000 years, provide analogous repetitions of glacial cycles. DOE built confidence in the selection of the first 10,000 years of the cycle as an analog for the next 10,000 years at Yucca Mountain by showing that the last 400,000-year cycle was similar to the previous 400,000-year cycle (400,000 to 800,000 years ago).

DOE constructed past climates for particular glacial stages (SAR Section 2.3.1.2.1.2.4) using the Forester, et al. (1999aa) analysis of ostracode occurrences in lake sediment obtained from composite core OL-92, drilled in 1992 at Owens Lake, California, together with observed flows in Owens River for wet years. Ostracodes are microfossils, with different species having different environmental preferences for salinity and temperature. In DOE's analysis, the ostracode-based salinity of the paleo-Owens Lake serves as a proxy for annual precipitation.

DOE used the abundance of five different species to infer compatible climatic parameters of temperature and seasonality and then developed future-climate parameters from meteorological stations with long records in locations where those species currently exist. DOE built confidence in the climate estimators using diatom records from the same core samples.

DOE's procedure yielded three representative climates for the first 10,000 years after closure: (i) modern (present-day) climate for the first 600 years, (ii) monsoonal climate for 1,400 years, and (iii) glacial transitional climate for the remaining 8,000 years. Relatively speaking, these three climates can be described as hot and dry, hot and wet, and cool and wet, although all are classified as arid or semi-arid climates. DOE calculated sample-average mean annual precipitation values for the monsoon and glacial transition climates that were 1.59 and 1.63 times the sample-average mean annual precipitation for the present-day climate (SAR Tables 2.3.1-2 through 2.3.1-4), and nominal mean annual temperatures for the monsoon and glacial transition climates were 0.9 °C [1.6 °F] warmer and 5.5 °C [9.9 °F] cooler, respectively, than the nominal mean annual temperature for the present-day climate using values from SNL Tables F-22 through F-24 and Eq. F-47(a) (2007az). DOE's representation of a monsoonal climate also exhibited a shift in seasonality, with summer convection storms making up a larger fraction of its annual total precipitation than for either the present-day or glacial-transition climates.

The NRC staff notes that DOE provided a reasonable description of its approach of using orbital cycles covering the past 800,000 years integrated with available paleoclimatic data to develop the timing and duration of climates over the next 10,000 years. The NRC staff compared the DOE description with staff's understanding (e.g., Stothoff and Walter, 2007aa) of paleoclimatic data and approaches for projecting future climates based on paleoclimatic information. On the basis of this understanding, the NRC staff notes that DOE incorporated features and processes important for using paleoclimate reconstructions in projecting future climates. The paleoclimatic information DOE used to develop the timing and duration of climates over the next 10,000 years is reasonably representative of Yucca Mountain because DOE obtained the paleoclimatic data from the region near Yucca Mountain.

DOE considered the two primary uncertainties in the data sets used to forecast future climate at Yucca Mountain to be the standard deviation associated with the Devils Hole ages and the uncertainty of the timing of climate change implied by the Devils Hole record (SAR Section 2.3.1.2.2.1.4). DOE also considered the two primary uncertainties in model forecasts of future climates to include (i) uncertainty in the location of the past–present equivalency point in the Owens Lake record, (ii) uncertainty arising from the chaotic nature of the climate system, and (iii) uncertainty in selecting a particular past climate sequence to forecast the future (SAR Section 2.3.1.2.3.2).

From the DOE-identified primary uncertainties and the downstream uses of the DOE future-climate model, NRC staff identifies three specific aspects of the DOE future-climate model where uncertainties may have a potential effect on repository performance: (i) uncertainty in timing and duration of climate states used for performance assessment calculations, (ii) uncertainty in climatic conditions during the post-thermal-pulse period when temperatures near drifts drop below boiling (dominated by the glacial-transition climate state in the DOE future-climate model), and (iii) uncertainty in climatic conditions from anthropogenic activities.

Uncertainty in Timing and Duration of Climate States

DOE used three climate states, each with a constant climate, to represent millennial-scale temporal variations found in the paleorecord. Uncertainty in the timing of transitions between the projected climate states could potentially impact estimates of future infiltration, which in turn, may affect unsaturated-zone flow, seepage rates, thermal histories, and radionuclide transport. The NRC staff notes that the timing for the transition between the monsoon and glacial-transition climate states has a minor impact on performance assessment results for the following reasons:

- The mean annual precipitation values for these two climate states are similar. DOE calculated sample-average mean annual precipitation values for the monsoon and glacial-transition climate states that were 1.59 and 1.63 times the sample-average mean annual precipitation for the present-day climate state (SAR Tables 2.3.1-2 through 2.3.1-4).

- The mean annual infiltration values for these two climate states are similar. DOE calculated weighted-average mean annual infiltration values over the repository footprint for the monsoon and glacial-transition climate states of 15.88 and 21.25 mm/yr, respectively, as shown in DOE Enclosure 1, Table 1 (2010ai). The weighted-average glacial-transition mean annual infiltration was 1.34 times larger than the weighted-average monsoon mean annual infiltration.

- The consequence of a much larger difference in mean annual infiltration has little consequence for performance. DOE stated that concurrently increasing areal-average mean annual infiltration by a factor of 2.39 for the monsoon state and 1.81 for the glacial-transition state has little effect on performance assessment results, as described in DOE Enclosure 5 (2009bo).

The NRC staff notes that the timing of the transition between the present-day and monsoon climate states, as set by DOE at 600 years following repository closure, has low consequence in performance assessment calculations for the following reasons:

- The present-day climate state predominantly corresponds to above-boiling conditions within emplacement drifts, and SAR Section 2.3.3.1 asserted that seepage into drifts is not expected to occur where rock above the repository exhibits above-boiling temperatures or dryout conditions. DOE estimated that above-boiling conditions within emplacement drifts may persist for several hundred to more than 1,000 years, depending on emplacement drift location (SAR Section 2.3.3.3.3.1).

- An early onset of a high-infiltration (monsoon or glacial transition) climate may reduce the duration of the thermal period, but it could not lengthen the duration of the post-thermal-pulse period affected by seepage by more than 600 years out of the first 10,000 years of performance.

- A delayed onset of a high-infiltration climate after the present-day climate state would result in smaller estimates of mean annual infiltration than are incorporated in the performance assessment.

The NRC staff notes that the potential for an early transition to a full-glacial climate state has low probability of occurring during the first 10,000 years after closure and low consequence if the transition occurs after 10,000 years. This is based on DOE's estimated return to a full-glacial climate state 30,000 years (SAR Section 2.3.1.2.1.2.3) after permanent closure. On the basis of on the paleoclimatic data from the region around Yucca Mountain, it is unlikely that the full-glacial climate would occur during the first 10,000 years after permanent closure. Furthermore, an earlier return of the full-glacial climate up to 20,000 years sooner than the DOE prediction would not change flux rates, because DOE used the prescribed deep percolation flux rate for the post-10,000-year period.

The glacial-transition climate represents 8,000 years of constant climate in the DOE model. Because millennial-scale fluctuations in climate are reflected in the paleo-records, NRC staff evaluated this model using a constant climate for 8,000 years instead of an alternative representation reflecting millennial-scale variations in climate. The NRC staff notes that the use of millennial-average net infiltration rates in performance assessment calculations, rather than the constant climate that DOE used for the final 80 percent of the 10,000-year performance period, has low consequence for performance for the following reasons:

- DOE included extreme events in developing the millennial-average net infiltration rates. DOE included expected variability in annual precipitation over a 1,000-year period in the net infiltration calculations (SAR Section 2.3.1.3.3) (e.g., including the calculated wettest year within each 1,000-year sequence).

- Using DOE information from its performance assessment sensitivity analyses, NRC staff notes in TER Section 2.2.1.3.6.3.2 that DOE's performance assessment results would not be significantly affected by including temporal variability of percolation about the long-term-average percolation flux within a climate state into performance assessment calculations. Because net infiltration is closely linked to percolation, DOE performance assessment calculations would not be strongly affected by fluctuations in climate that lead to infiltration flux varying about the long-term-average infiltration flux.

In summary, DOE adequately represents uncertainty in the timing and duration of climate states for performance assessment because DOE showed that changes in the representation for the timing and duration of the climate states have a low consequence for performance assessment calculations.

Uncertainty in Climatic Conditions During the Post-Thermal-Pulse Period

DOE used the glacial-transition climate state to represent climate during the final 80 percent of the first 10,000 years of repository performance. Of the three climate states that DOE used within the first 10,000 years of performance, the NRC staff considers the glacial-transition climate state to have the largest potential for affecting repository performance because this climate state has the longest duration, and seepage into emplacement drifts is least affected by the thermal pulse during this climate state. During the thermal pulse, above-boiling conditions and evaporation reduce the flux of water reaching drifts (SAR Section 2.3.3.1). In evaluating the DOE approach to representing uncertainty in the glacial-transition climate state, the NRC staff considers (i) methodology that DOE used to estimate future climatic conditions, (ii) parameters that DOE found contribute most heavily to uncertainty in downstream applications, (iii) available estimates for climate change from the last glacial maximum, and (iv) representation of intermediate climate fluctuations using a constant climate for a climate state.

DOE first considered the presence and absence of indicator species within Owens Lake to infer changes in climatic conditions relative to present-day climate, then estimated compatible climatic conditions from present-day locations where the same indicator species currently exist. DOE represented uncertainty regarding climatic conditions using upper and lower bounds for mean annual precipitation and mean annual temperature for each climate state. To account for the uncertainties in translating paleoclimatic indicators into meteorological records, DOE selected several meteorological stations to represent each of the bounding climatic conditions at an elevation of 1,524 m [5,000 ft] at Yucca Mountain (SAR Section 2.3.1.2.3.1.2). The criteria for selecting present-day meteorological stations, outlined in SAR Section 2.3.1.2.3.1.2, include (i) presence of Owens Lake indicator species, (ii) mean annual temperature, (iii) rain-shadow effects, (iv) position of the polar front, and (v) length of observational record. Further, DOE selected meteorological stations such that (i) the climate states have larger mean annual precipitation values for upper bounds than for lower bounds and (ii) the upper-bound mean annual precipitation values are larger than the present-day observations for both the monsoonal and glacial transition climate states.

DOE concluded that mean annual precipitation is one of the two parameters that control uncertainty in MASSIF estimates of net infiltration for all climate states (SAR Section 2.3.1.3.3.2.2). Further, DOE based the selection of representative meteorological stations on the station record length and on observations that are sensitive to mean annual temperature and precipitation seasonality (i.e., ostracode species) without using criteria based on specific values of mean annual precipitation. Uncertainty in mean annual precipitation is a dominant source of uncertainty in the DOE net infiltration estimates, and the DOE procedure for selecting representative meteorological stations yields relatively large uncertainty in mean annual precipitation.

DOE selected meteorological stations for the glacial-transition climate with average observed annual precipitation between 207 and 241 mm/yr [8.1 and 9.5 in/yr] for the lower bound and between 419 and 455 mm/yr [16.5 and 17.9 in/yr] for the upper bound (SAR Table 2.3.1-6). For comparison, meteorological stations at Yucca Mountain have observed mean annual precipitation between 183 and 213 mm/yr [7.2 and 8.4 in/yr], averaging 199 mm/yr [7.8 in/yr]. Accordingly, the mean upper- and lower-bound mean annual precipitation estimates for the glacial-transition climate state are approximately 2.2 and 1.2 times the average observed present-day precipitation of 199 mm/yr [7.8 in/yr] at Yucca Mountain.

DOE does not expect that a full glacial climatic state would occur within the next 30,000 years, and the SAR did not estimate climatic conditions for a full glacial climatic state. NRC staff has nonetheless reviewed published estimates for mean annual precipitation during the last glacial maximum in the region surrounding Yucca Mountain, as detailed in Stothoff and Walter, Section 2.2 (2007aa). Several of the published estimates listed in Stothoff and Walter (2007aa) quantitatively considered the effects of mean annual precipitation on a water balance. Such quantitative estimates inferred changes in mean annual precipitation and mean annual temperature by considering elevation changes for plant species that have known environmental preferences, hydrologic balances for paleolakes, extent of glacial advances, and regional groundwater balances. Among these estimates, the largest estimated value for mean annual precipitation at the last glacial maximum suggests that mean annual precipitation was 1.9 times larger than present-day mean annual precipitation at Yucca Mountain.

The NRC staff notes that DOE reasonably represents uncertainty in the magnitude of mean annual precipitation change during the glacial-transition climate state. The basis for this is that the upper bound of mean annual precipitation values that DOE used to represent the upper bound of mean annual precipitation during the glacial-transition climate state is substantially larger than published quantitative estimates for mean annual precipitation during the last glacial maximum in the region surrounding Yucca Mountain. The NRC staff further notes that DOE reasonably represents mean annual temperature and precipitation seasonality during the glacial-transition climate state because DOE based the values on indicators in the region surrounding Yucca Mountain that are sensitive to these factors.

Uncertainty in Climatic Conditions From Anthropogenic Activities

DOE stated that the predicted modern climate is based on “climate records that implicitly include effects of modern society over the duration of historical record” SNL, Section 6.2, FEPs 1.4.01.00.0A and 1.4.01.02.0A (2008ab) and DOE, Enclosure 8 (2009cr). Uncertainty in the incorporation of anthropogenic effects on climate predictions used as input for net infiltration estimates is twofold. First, monsoonal and glacial-transition climate analog sites are derived from interpretation of the paleoclimatic record (e.g., Owens Lake ostracode and diatom observations). However, current levels of greenhouse gases (i.e., dominantly CO_2 but including other gases) are elevated beyond any levels indicated in paleoclimate records covering the past 800,000 years. Second, the effect of the global climatic changes on Yucca Mountain climate is uncertain. To address these uncertainties, DOE described consequences to infiltration estimates caused by likely projections of climate change in the desert Southwest considering anthropogenic influences.

In DOE Enclosure 8 (2009cr), DOE considered projected climate changes in the desert Southwest, described by the International Panel on Climate Change (Christensen, et al., 2007aa) to assess potential consequences of anthropogenic climate change on repository performance. Projected regional climate change estimates indicate the desert Southwest is likely to see temperature increases that are higher than average global warming and annual precipitation that is likely to decrease in the next century. DOE, Enclosure 8 (2009cr) described the projected regional climate changes as having potential consequences, including

- Improved repository performance under warmer and drier conditions, because warmer temperatures and decreased precipitation lead to decreased net infiltration

- Insignificant effects on repository performance under a warmer and wetter climate or early onset of monsoon conditions induced by anthropogenic climate change, because most of the repository would be above boiling during the first 600 years after closure [DOE considered climate changes Christensen, et al. (2007aa) projected to be similar to but smaller than that represented by the shift from modern climate to the monsoonal state]

- Improved repository performance if anthropogenic climate change caused a delay in the onset of the glacial-transition climate because net infiltration under the cool and wet glacial-transition climate state is higher than would occur for earlier climate states.

The NRC staff notes that DOE adequately bounded the effects of anthropogenic climate change in its performance assessment calculations because DOE showed that (i) net infiltration is not consequential to repository performance within the first 600 years after repository closure and (ii) net infiltration under credible projected climate changes would be overestimated by using the climate states already used for performance assessment calculations.

Summary of Conclusions Regarding Climate Change for the Next 10,000 Years

In summary, DOE adequately represents future climate uncertainty because (i) DOE provided a reasonable description of the approach and data used to represent climate states, and furthermore, that those climate states and their uncertainties are representative of Yucca Mountain during the first 10,000 years after closure, (ii) the DOE representation of the timing and duration of the climate states has a low consequence for performance assessment calculations, (iii) DOE used upper-bound values for mean annual precipitation during the glacial-transition climate state (post-thermal pulse) that are more extreme than the largest available published estimate for the last glacial maximum, and (iv) DOE projected changes to climate stemming from anthropogenic activities that are either not consequential to repository performance or are bounded by the climate states used for performance assessment calculations.

2.2.1.3.5.3.2.2 Local Spatial and Temporal Variation of Meteorological Conditions

In this section, NRC staff evaluates the DOE model for climatic and meteorological conditions during each climate state (i.e., climate conditions at short time scales). This section addresses input data characterization and uncertainty.

DOE represented meteorological conditions for each climate state using sampled 1,000-year sequences of daily estimates for total precipitation and temperature extremes, representing conditions at a reference elevation of 1,524 m [5,000 ft]. The MASSIF infiltration model subsequently estimates precipitation and temperature variability over each day using the daily values. DOE represented spatial variation by projecting the daily precipitation and temperature values to the infiltration-model cells using elevation-dependent lapse rates. DOE considered precipitation rates with up to a 1,000-year recurrence period in generating the 1,000-year sequence. DOE selected 10 representative 1-year sequences out of each 1,000-year sequence to estimate long-term-average net infiltration. DOE used a water year representation, initiating each 1-year sequence on October 1 to capture the cycle of winter precipitation and large summer potential evapotranspiration. Each simulation was initiated with conditions representing extended summer evapotranspiration. DOE stated that the wettest years were sampled to ensure the disproportionate influence of wet years was captured for net infiltration estimates.

NRC staff reviewed DOE's representation of spatial and temporal variability in meteorological parameters for estimating net infiltration in the two following sections.

Spatial Variability in Meteorological Parameters

DOE considered the effect of elevation on meteorological parameters by adjusting the estimated daily values for precipitation and temperature extremes according to regional patterns in mean annual precipitation and mean annual temperature. The NRC staff notes that the dependency of meteorological parameters on elevation provides the only spatial variability of those parameters across the site. NRC staff notes, however, that DOE's approach of representing spatial variability in precipitation and temperature for the purpose of providing long-term-average net infiltration estimates as a boundary condition to the site-scale unsaturated zone flow model is reasonable for the following reasons:

- The size of the relevant washes is small relative to the size of typical precipitation events. Precipitation patterns during individual precipitation events are likely to be relatively spatially uniform within a few kilometers [a few miles], especially during large storms and the frontal storms that are predominant in the cooler periods of the year with small potential evapotranspiration. For example, maximum observed 24-hour precipitation ranged between 78.5 and 87.1 mm [3.1 and 3.4 in] (SAR Table 1.1-23) for the 10 meteorological stations within 5 km [3 mi] of the repository footprint in the largest 24-hour event ever recorded with onsite meteorological stations (September 21 to 22, 2007). The repository footprint has an area of 4.6 km^2 [1.8 mi^2] and includes more than a dozen washes. The NRC staff notes, on the basis of the smaller size of individual washes compared to the larger size of large frontal storms, that precipitation patterns are likely to be relatively spatially uniform within individual washes above the proposed repository footprint.

- To estimate long-term average net infiltration, DOE assumes the washes are hydrologically independent (i.e., there is no lateral flow between washes). On the basis of site topography and drainage system (SAR Figure 1.1-5), the NRC staff notes that the washes within the repository footprint can reasonably be considered hydrologically separate.

- To estimate long-term average net infiltration, DOE assumed a given meteorological sequence is equally applicable to each wash in the model domain. This is a reasonable assumption because each small wash is likely to exhibit similar frequencies of meteorological patterns over long periods of time due to close proximity.

- DOE cited regional studies in SNL Sections 6.4.11 and 6.4.5.3 (2007az) indicating that mean annual precipitation and mean annual temperature are correlated with elevation even though local topography can modify the relationship. DOE derived the precipitation lapse rate used for infiltration calculations from meteorological stations in the Yucca Mountain region, as outlined in SNL Appendix F.2 (2007az). The NRC staff confirmed that the DOE precipitation lapse rate is comparable to other regional relationships, within the bounds of uncertainty, over an elevation difference typical of the repository footprint. Because DOE showed that the infiltration model results, other model results, and observational evidence from other locations all exhibit a systematic trend of larger net infiltration as mean annual precipitation increases (SAR Section 2.3.1.3.4), the NRC staff notes that DOE reasonably considers the systematic elevation-dependent variation in mean annual precipitation on net infiltration for calculating long-term average net infiltration.

- DOE derived the temperature lapse rate used for infiltration calculations from a textbook value for the dry adiabatic lapse rate, representing an upper bound representation, as detailed in SNL Appendix C.1.4 (2007az). Although the temperature lapse rate may overestimate the regional lapse rate, DOE showed in SNL Section 7.1.4 (2007az) that the infiltration model results are not sensitive to the temperature lapse rate. Thus the temperature lapse rate for calculating long-term-average net infiltration is reasonable.

In summary, the DOE approach of spreading a single, generated meteorological time history throughout the model domain using elevation-dependent lapse rates for calculating boundary condition fluxes for the site-scale unsaturated zone flow model is reasonable. This approach is unlikely to systematically bias the calculated areal-average long-term-average net infiltration. This is further based on the observations that (i) the individual washes within the repository footprint have relatively small areas compared to typical storms, (ii) the individual washes within the repository footprint are hydrologically independent with respect to lateral flow, and (iii) systematic trends in the meteorological parameters that most strongly affect net infiltration are incorporated in the infiltration calculations.

Temporal Variability in Meteorological Parameters

DOE represented temporal variability of meteorological conditions using sampled daily values for precipitation, temperature minimum and maximum, and wind speed. On days with precipitation, DOE subdivided the daily calculation into two parts, representing storm and nonstorm conditions, and used wet-day instead of dry-day temperature values. DOE described the statistical parameters characterizing these meteorological components as varying sinusoidally over the year. DOE considered the representation of temporal variability adequate because measured regional and Yucca Mountain site data were used to develop precipitation and temperature sequences.

The NRC staff notes that DOE's representation of cool-season (winter) precipitation is a risk-significant aspect of temporal variability of meteorological conditions, because warm-season precipitation has a disproportionately small effect on net infiltration. This is based on the NRC staff's literature surveys and independent confirmatory investigations in Stothoff and Walter, Section 3.6 (2007aa) and Stothoff (2008aa). In particular, the NRC staff notes measurements and analyses indicate that approximately 10 percent of recharge at Mount Charleston (in the Spring Mountains, southeast Nevada) has the isotopic signature of summer precipitation, which represents approximately 30 percent of the annual precipitation (Winograd, et al., 1998aa).

Accordingly, the NRC staff compared DOE's mathematical representation of precipitation in SNL Appendix F (2007az) with summary observations from meteorological stations DOE used to represent mean winter and summer precipitation for potential future climate states. In the analysis (Stothoff, 2010aa, Section 3), the NRC staff noted that several statistical properties from the observed precipitation data sets fell close to statistical properties of DOE's precipitation representation. The NRC staff recognizes that there is uncertainty in estimating mean annual precipitation from observations; for example, average precipitation totals from 1994–2006 for five Yucca Mountain Project meteorological stations, reported in SAR Tables 1.1-10 through 1.1-12, 1.1-15, and 1.1-18, differ on average by approximately 7 percent from values for 1993–2004 reported in SNL Table 6.1-4 (2006aa). The NRC staff notes that DOE's representation of precipitation is reasonable for calculating daily precipitation for net infiltration because the statistical model has seasonal patterns for precipitation values comparable to observations.

The NRC staff compared DOE's representation for temperature as sinusoidally varying during the year to observations from meteorological stations in Nevada, Utah, California, and Arizona, as described in Stothoff Figures 5-8 and 5-09 (2008aa). The amplitude and seasonality DOE represented is comparable to the observations. The NRC staff considers net infiltration calculations to be relatively sensitive to temperature on days with precipitation because evaporation during precipitation affects the amount of water infiltrating during an event. Calculations of net infiltration, however, are relatively insensitive to temperature fluctuations on days without precipitation because it typically takes weeks for evapotranspiration rates to remove the soil moisture from a large storm. The NRC staff notes that DOE considered temperature in daily meteorological sequences because DOE used a representation for temperature that has amplitude and seasonality comparable to observations, and because DOE considered separate parameterizations for wet and dry days.

2.2.1.3.5.3.3 Net Infiltration

In this section, NRC staff evaluates DOE's model for net infiltration during the 10,000 years following repository closure. The NRC staff evaluates the downstream uses of the net infiltration results and the effect of uncertainty in net infiltration on DOE performance assessment calculations. The focus of the NRC staff's assessment is on those aspects of DOE's net infiltration model that are most important for repository performance.

In evaluating repository performance with respect to unsaturated flow (see TER Sections 2.2.1.3.6.3.2 and 2.2.1.3.6.3.4), the NRC staff identified systematic changes in seepage as the dominant performance-affecting consequence of spatial and temporal variability in percolation fluxes within the unsaturated zone below the Paintbrush Tuff nonwelded (PTn). Also in those TER sections, the NRC staff noted that the DOE model is relatively insensitive to other aspects such as spatial variability in deep percolation, local flow focusing, decadal-to-centennial climatic variability, episodic deep percolation pulses, and calibration of net infiltration uncertainty. Because seepage and percolation closely track net infiltration, the NRC staff considers areal-average net infiltration the dominant performance-affecting feature of the infiltration model with respect to DOE's performance assessment calculations.

DOE also used the areal-average net infiltration results in performance assessment calculations related to the saturated zone, adjusting saturated zone groundwater fluxes for future, wetter climates using net infiltration (SAR Section 2.3.8). The NRC staff notes, however, that repository performance is more sensitive to changes in seepage than changes in saturated zone groundwater fluxes. Seepage directly affects radionuclide releases, so uncertainty in seepage directly affects the uncertainty in calculated dose. Uncertainty in groundwater flux rates has a much smaller effect on uncertainty in dose calculated by the DOE performance assessment because (i) nonsorbing radionuclides (e.g., Tc-99) are transported through the saturated zone in a small fraction of the performance period regardless of uncertainty in groundwater flux rates and (ii) uncertainties in transport rates for sorbing radionuclides are dominated by uncertainties in aspects other than groundwater flux, such as sorption characteristics. The following subsections contain the NRC staff evaluation of the technical bases for DOE's models and inputs used to estimate areal-average net infiltration.

Submodels for Net Infiltration

This subsection addresses model integration, justification, and uncertainty.

DOE used a water balance approach to integrate processes and features acting at and near the ground surface, to a depth at the bottom of the root zone. A water balance approach requires that the supply of water (precipitation and run on) at any location is equal to the sum of other components (e.g., evapotranspiration, change in water storage, runoff, and net infiltration). In the water balance, uncertainty in precipitation and evaporation, the largest components, can affect estimation of the much smaller component of net infiltration. DOE described the development and integration of features and processes into conceptual and numerical models in SAR Sections 2.3.1.3.1 and 2.3.1.3.2 and SNL Sections 6.3 and 6.4 (2007az). DOE separates the water balance model into the key MASSIF elements of

- Climate and meteorology, using daily precipitation, temperature, and snowmelt

- Subsurface water movement and storage, using a one-dimensional vertical soil water balance

- Surface runoff and run on, using topography-based flow routing

- Evapotranspiration, using the FAO-96 approach (Allen, et al., 1998aa) modified for natural vegetation

- Reference evapotranspiration, using the FAO Penman-Monteith method (Allen, et al., 2005aa)

MASSIF comprises linked submodels for the identified processes and routines to manipulate geographically distributed input data into the formats required for the calculations. The geographically distributed input data, which are defined at each 30-m [98-ft] pixel across the Yucca Mountain area, include soil and rock hydrologic properties, topography, vegetation factors, and climate information (e.g., precipitation and temperature).

In MASSIF, net infiltration is defined as the water that moves below the active zone where evaporation and plant uptake are significant processes. DOE assumed that the active zone does not penetrate the bedrock, so that water passing into the bedrock becomes net infiltration. DOE considered this assumption to be conservative with respect to the magnitude of net infiltration, because plant roots, especially in areas with thin soil cover, develop in bedrock fractures and thus would reduce net infiltration by taking up water for transpiration.

DOE described in SNL Section 6.2.3 (2007az) six criteria used for selecting the infiltration model components: (i) the model and components should be consistent with the overall project purpose, (ii) the model component complexity should be consistent with the available input data, (iii) model components should be consistent with other model components, (iv) the model should be computationally efficient, (v) the model should be accessible and open, and (vi) the model and model components should demonstrate reasonable predictive capability. DOE described cases in the literature where the algorithms and approaches in submodels have been utilized at other locations in semi-arid areas. DOE explained that these criteria are motivated by the large spatial and temporal scales being modeled, the limited objectives of the infiltration model, and the need for numerous simulations to assess sensitivities and address multiple climate scenarios. DOE further explained that the infiltration model is not intended to describe the detailed spatial and temporal character of water movement.

The NRC staff has experience in evaluating the features, processes, and models used for arid zone hydrology gained from two decades of interactions with DOE and from documenting the NRC staff's independent modeling (Stothoff, 2008aa; Stothoff and Walter, 2007aa). On the basis of knowledge gained through this experience, the NRC staff notes that DOE reasonably described the technical basis for the infiltration conceptual model and the associated mathematical model in SAR Section 2.3.1.3 and supporting documents. Furthermore, the MASSIF submodels are reasonable for their intended use because the algorithms and approaches (i) are used in the scientific community, (ii) are reasonable for the spatial and temporal scales described in DOE's six criteria for the infiltration model (see previous paragraph for DOE's criteria), and (iii) consider downstream uses. The DOE submodels are consistent with staff's experience regarding approaches for modeling net infiltration in arid environments (i.e., Stothoff and Musgrove, 2006aa; Stothoff, 2008aa).

To build confidence in the water balance approach for one-dimensional water storage and movement, DOE compared results with both measured data and results from an alternative model solving the Richards equation for unsaturated flow (SAR Sections 2.3.1.3.4.1 and 2.3.1.3.4.2). The Richards equation includes capillary effects, unlike the water balance approach in MASSIF. The DOE comparisons focused on the MASSIF submodels for water storage, evapotranspiration, and one-dimensional vertical movement of water. The measured lysimeter data were from two locations, the Reynolds Creek Experimental Watershed in New Mexico and the Nevada Test Site. DOE also compared MASSIF results with results from the Richards-equation-based models for one-dimensional, stylized problems with varying soil and plant root depths. The NRC staff reviewed the DOE comparisons of MASSIF and Richards-based-equation model results with measured data and the stylized one-dimensional problem. On the basis of this review, the NRC staff notes that the water storage and evapotranspiration submodels in MASSIF adequately represent both the measured data and the Richards-equation-based model results. The staff notes that the MASSIF and the Richards-equation-based model have similar responses, and that the two numerical models track the observations to a quantitatively similar degree of accuracy.

In another comparison with measured data, DOE contrasted MASSIF results against measurements from streamflow gauges in three subwatersheds at Yucca Mountain for several storm events, as outlined in SAR Section 2.3.1.3.4.1 and SNL Section 7.1.3 (2007az). SAR Figure 2.3.1-46 illustrated that the timing and magnitude of measured and modeled runoff are reasonably well matched with a particular set of input properties that lie within the uncertainty range considered in the net infiltration model. The input properties used to match streamflow observations represent the nominal properties with soil hydraulic conductivity adjusted to increase upland runoff and enhance channel infiltration. DOE indicated that local variations within the watersheds may have also been a factor in the comparison. On the basis of the adjustments DOE made to match streamflow observations, the uncertain input parameter distributions used for performance assessment calculations may, in the NRC staff's view, create a bias toward overestimating the fraction of MASSIF-calculated total net infiltration that upland infiltration contributes relative to channel infiltration.

The NRC staff recognizes, however, that uncertainty in runoff only affects the distribution of infiltration and does not affect the areal-average infiltration; it is this latter factor that is of greater importance to repository system performance. Thus, the surface runoff submodel provides a reasonable basis for predicting runoff over the entire repository for the next 10,000 years because (i) the surface runoff submodel algorithms are commonly used; (ii) DOE showed that, with appropriate input parameters, the MASSIF surface runoff submodel is capable of providing a reasonable match to observed runoff during storm events within representative subwatersheds; (iii) the DOE results are comparable to an independent, alternative model for runoff and infiltration (Woolhiser, et al., 2000aa, 2006aa); (iv) the uncertainty in net infiltration stemming from uncertain runoff is small; and (v) the DOE performance assessment calculations are not sensitive to different representations of spatial variability in net infiltration, as described in DOE, Enclosure 4 (2009cr). A more general evaluation of uncertainty in spatial variability is in a subsequent section of TER 2.2.1.3.5.3.3 entitled "Net Infiltration Results."

Input Parameters

This subsection addresses characterization of data and propagation of data uncertainty.

NRC staff understands that most net infiltration within the proposed repository footprint occurs in shallow soil as a pulse over a few days to weeks following large precipitation events during periods of low potential evapotranspiration. This understanding is based in part on the DOE documentation of its infiltration modeling. Further in this regard, the NRC staff has experience with infiltration processes and modeling at Yucca Mountain (Stothoff, 2008aa, and references therein). During the short intervals with large net infiltration pulses, which dominate the long-term-average net infiltration, the flux of water passing into the bedrock in shallow soil can dominate evapotranspiration in the water balance. Accordingly, the NRC staff focused its review in this section on aspects of the site affecting rapid transmission of pulses to bedrock in shallow soil; in particular, soil water storage.

Available soil water storage during a precipitation event depends on soil depth, soil water-holding capacity, and the antecedent soil moisture content (i.e., how dry the soil column is prior to the event). Soil and bedrock hydraulic properties affect the rate of soil water movement toward and into the bedrock. In DOE's representation, water drains into the bedrock once the water storage capacity of the overlying soil layers is exceeded, thereby avoiding evapotranspiration. The drainage rate into the bedrock is controlled by the layer, soil or bedrock, with the smaller value bulk permeability. Using analyses reported in SNL Section 7.1.4 (2007az), DOE stated that net infiltration estimates are not sensitive to uncertainty in bulk bedrock permeability, in part because of the limited spatial extent where bedrock controls the drainage rate. Therefore, NRC staff's review focuses on the soil depth, soil water-holding capacity, and changes to hydrologic input properties in future climates. Soil water-holding capacity depends on the soil unsaturated hydraulic properties.

Areas with thin soil cover are particularly important because DOE identified infiltration as most readily occurring in shallow soil. Areas with shallow, or thin, soils comprise 70 percent of the unsaturated model domain (Soil Depth Class 4, SAR Section 2.3.1.3.2.1.3) and appear to cover an even larger fraction of the repository footprint (SAR Figure 2.3.1-19). In DOE Enclosure 5 (2009cr), DOE identified soil depth in areas with shallow soil cover as the most important hydrologic property input for the infiltration model, with model results approximately as sensitive to uncertainty effective soil depth as uncertainty in precipitation. The effective soil depth is defined as the single soil depth value that, if applied everywhere, yields the same areal-average net infiltration as the actual soil-depth distribution.

Soil Depth Class 4 represents soil depths between 0 and 0.5 m [0 and 1.6 ft] and corresponds to eolian deposits with various mixtures of entrained rock on hillslopes and ridgetops. DOE sampled a single sampled effective soil depth value to characterize Soil Depth Class 4 for each realization and assigned the value to every grid cell representing that class for the corresponding simulation. DOE used two datasets to support its effective soil depth distribution for this class: (i) 35 site observations recorded as point measurements ranging from depths of 0 to 3 m [0 to 9.8 ft], with a recommended median of 0.25 m [0.82 ft], as described in SNL Table 6.5.2.4-2(a) (2007az), and (ii) 8 site observations recorded as general site characteristics at locations such as drill pads. DOE described the measurements as approximately lognormally distributed and, on the basis of geometric and arithmetic means of the two sets of observations, derived bounds on effective soil depth ranging from 0.1 to 0.5 m [0.33 to 1.6 ft]. DOE also analyzed 56 NRC soil depth measurements (Fedors, 2007aa) obtained from site visits focusing on the thin soils of the east-trending upper washes over the southern half of the repository footprint. DOE described the available NRC staff's measurements as approximately following a lognormal distribution.

For performance assessment calculations, DOE described the effective soil depth in Soil Class 4 as equally likely for any value between the upper and lower bounds. In selecting the uniform statistical distribution, DOE considered the difficulty in measuring soil depth, uncertainty in the mean of the observations, and uncertainty in how soil depth affects net infiltration, as detailed in DOE, Enclosure 5 (2009cr). In the same document, DOE stated that sensitivity results in SNL Appendix H (2007az) indicate that calculated areal-average net infiltration is approximately linearly dependent on effective soil depth for Soil Class 4, and that shallow soil depths are not underrepresented in the effective soil depth distribution for Soil Depth Class 4. From this, DOE concluded that use of a uniform distribution for effective soil depth does not underestimate average net infiltration.

NRC staff notes that DOE adequately characterized the statistical properties of the observations for use in the net infiltration modeling from two perspectives. First, DOE provided reasonable bounds on the uncertainty in effective soil depth in Soil Depth Class 4. A shallower extreme would require that more than half of the area with shallow soil has a soil depth less than 0.1 m [0.33 ft], which, in contrast to observations, implies extensive exposures of bare rock in the infiltration model domain; a deeper extreme would reduce net infiltration. Second, the NRC staff's confirmatory field investigations (Stothoff, 2008ab) indicate that comparable topographic locations on hillslopes and ridgetops have similar soil depths across the repository footprint, consistent with the DOE description of an eolian source for the fine component of the soil.

NRC staff considered the effect of uncertainty in the soil depth distribution for Soil Depth Class 4 on the estimate of net infiltration. As described in SNL Table 6.5.2.4-2(a) and Section 7.2.4(a) (2007az) and DOE, Enclosure 5 (2009cr), DOE used a uniform distribution, while noting that the measured data may better fit a lognormal distribution. To assess the DOE representation of soil depth uncertainty, the NRC staff used DOE's sensitivity analyses for fixed aleatory uncertainty under present-day and glacial transition climate states, as outlined in SNL Figures H-3, H-4, H-11, and H-12 (2007az), to estimate the consequence of decreasing median soil depth. Using this alternative representation for how uncertainty might be distributed, the NRC staff's analysis suggested areal-average net infiltration would increase by 43 to 61 percent under the present-day climate state and by 29 to 38 percent under the glacial transition climate state if mean soil depth decreased from 0.3 m to 0.2 m [0.98 ft to 0.65 ft] for Soil Depth Class 4 (Stothoff, 2010aa, Section 3). DOE, Enclosure 5 (2009bo) showed that increasing average net infiltration over the ambient flow model domain by percentages greater than 61 percent, due to uncertainty in the probability weights of the Generalized Likelihood Uncertainty Estimation methodology, did not strongly affect DOE's performance assessment calculations. Because the increase (29 to 61 percent) in net infiltration calculated by the NRC staff using an alternative representation of the uncertainty of soil depth distribution is significantly smaller than the increase DOE estimated for uncertainty in net infiltration due to the Generalized Likelihood Uncertainty Estimation weights, the NRC staff notes that uncertainty in soil depth distribution of Soil Depth Class 4 is also not important to performance. The NRC staff notes that DOE reasonably represents effective soil depth in Soil Depth Class 4. This is because the bounds on the uncertainty distribution for net infiltration are sufficiently wide to cover the effects of uncertainty in soil depth.

Water-holding capacity is calculated from soil hydraulic properties (i.e., porosity and water retention characteristics). DOE utilized a pedotransfer function derived for Hanford soils to relate Yucca Mountain soil texture to hydraulic properties for each soil group in the infiltration model domain. In the MASSIF model, larger water-holding capacity values result in smaller values of net infiltration. Because the hydrologic property relationship to soil texture may be different for soils from Hanford, Washington, compared to that for soils at Yucca Mountain, DOE, Enclosure 6 (2009cr) compared water-holding capacity used in MASSIF with two estimates made for local Yucca Mountain soils. The first set covers soils in Nye County, and the second set covers soils from the Yucca Mountain area but not used in the MASSIF model.

DOE, Enclosure 6 (2009cr) stated that the estimates of water-holding capacity used in MASSIF are smaller than those estimated for the other two sets of soils. On the basis of its review of this information, NRC staff notes that the development of average water-holding capacity values for soils from the Hanford-based pedotransfer function does not lead to underestimation of water-holding capacity for Yucca Mountain soils.

NRC staff considered the performance consequence of the DOE assumption that all geologic and geographic parameters in the infiltration model remain the same over the transition from dryer to wetter climates during the next 10,000 years. DOE Enclosure 2 (2009cr) described changes to soil depth, soil hydraulic properties, and bulk bedrock permeability under future climates that may include (i) greater chemical soil profile development and enhanced weathering of bedrock at the interface with soil, (ii) relatively larger soil depths and different soil depth distributions, and (iii) relatively smaller amounts of caliche filling bedrock fractures at the soil/bedrock interface. In DOE Enclosure 2 (2009cr), DOE described the potential consequences as either inconsequential or beneficial to repository performance:

- Projected changes to soil depth and soil hydraulic properties would tend to reduce the estimates of net infiltration. The NRC staff notes that DOE Enclosure 2 (2009cr) description of projected changes is consistent with current understanding of geomorphic responses to climate change in the desert Southwest [e.g., Bull, Section 2.5 (1991aa)].

- Where bedrock permeability values are greater than soil permeability, DOE considered the effect of a change in modeled bedrock properties as either inconsequential to net infiltration (if bedrock permeability increased) or to reducing infiltration (if bedrock permeability became smaller than the soil permeability). For more than half the ambient site-scale unsaturated zone modeling domain, DOE Enclosure 2 (2009cr) stated that bedrock permeability is greater than soil permeability.

- In the remaining area, DOE considered net infiltration as having a potential to increase only in the realizations where sampled bedrock permeability is smaller than soil permeability in the present climate, and only if bedrock permeability increases under a future climate. The NRC staff notes that this potential exists in less than half of the modeling domain for approximately half the realizations. The NRC staff calculated an upper-bound estimate for areal-average net infiltration that is 1.33 times larger than the DOE-calculated value, conservatively assuming that (i) half the area in half the realizations has low bedrock permeability and zero infiltration, (ii) all DOE-calculated net infiltration occurs in the remaining area with high bedrock permeability, (iii) increased bedrock permeability results in an upper bound of the areal-average net infiltration from the high bedrock permeability, (iv) the entire area has high bedrock permeability in all realizations, and (v) projected changes to soil depth and soil hydraulic properties do not result in lower net infiltration. The NRC staff notes that this upper-bound increase in areal-average net infiltration is less than the increase that DOE provided in DOE Enclosure 5 (2009bo) and is not significant to repository performance.

NRC staff notes that DOE's approach to maintain constant, but uncertain, soil depth and hydraulic properties and bulk bedrock permeability over the initial 10,000 years of performance is not likely to lead to consequential increases in areal-average net infiltration, because most expected changes to these properties would tend to reduce net infiltration in DOE's model. Further, DOE showed that upper-bound estimates for potential increases in net infiltration do not significantly change performance assessment results.

Net Infiltration Results

This subsection addresses propagation of uncertainty and support of model output for DOE's estimates of net infiltration for each climate state. Effects of net infiltration model results on repository performance are considered in how the infiltration model output is used in the unsaturated flow model, including seepage and unsaturated zone transport. NRC staff reviewed the net infiltration results in the context of average ratio of infiltration to precipitation, values of areal-average net infiltration, and spatial and temporal distribution of net infiltration.

DOE represents uncertain inputs to the MASSIF model with Monte Carlo sampling, using 40 realizations of selected hydraulic properties and climate characteristics to estimate net infiltration uncertainty for a climate state (SAR Section 2.3.1.3.3). For each realization, DOE calculated a process-level map of mean annual net infiltration by (i) creating a synthetic weather history representing 1,000 years, (ii) selecting 10 water years (a water year is October 1 through September 30 of the following year) out of the 1,000-year history to represent the range of dry to wet years, (iii) calculating total net infiltration for each water year using MASSIF, and (iv) averaging the 10 water-year net infiltration maps. DOE selected 4 of the 40 equally likely process-level mean annual net infiltration maps to represent the uncertainty in infiltration by (i) calculating areal-average net infiltration for each map, (ii) ranking the average values from low to high, and (iii) selecting the 4th, 12th, 20th, and 36th ranked map to represent the 10th, 30th, 50th, and 90th percentile ranking. The 12 maps of net infiltration are output provided to the unsaturated zone model for use as top boundary flux for the first 10,000 years. Because DOE used a standard approach for propagation of data uncertainty and conservatively assumed each stochastic realization was equally likely to occur, DOE reasonably propagated uncertainty in climate and hydrologic parameter inputs in development of the net infiltration maps.

DOE adjusted 4 of the 12 upper-boundary net infiltration maps developed for the first 10,000 years after closure to represent the probability distribution for deep percolation at the repository horizon for the post-10,000-year period. DOE selected the four upper-boundary net infiltration maps with the largest areal-average net infiltration within the repository footprint for the scaling procedure. The NRC staff reviews the procedure and technical basis for developing the post-10,000-year unsaturated zone model upper-boundary net infiltration maps in TER Section 2.2.1.3.6.3.2.

In NRC staff's analysis, three primary aspects of the response of net infiltration to climate may affect the performance of the proposed repository: (i) central tendency and uncertainty in areal-average time-averaged net infiltration, (ii) spatial variability in time-averaged net infiltration, and (iii) temporal changes in net infiltration. The NRC staff identified systematic changes in seepage as the dominant performance-affecting consequence of spatial and temporal variability in percolation fluxes below the PTn (TER Section 2.2.1.3.6.3.2). Through evaluation of DOE sensitivity analyses, the NRC staff notes in TER Section 2.2.1.3.6.3.2 that DOE's performance assessment results are not strongly affected by any of the following factors: (i) systematic changes in seepage arising from different representations of local flow focusing, (ii) long-term climatic variability, (iii) episodic infiltration pulses, or (iv) calibration of infiltration uncertainty. On the basis of this, the NRC staff also notes that spatial or temporal variability in net infiltration has little effect on performance assessment unless such variability results in a systematic change in areal-average net infiltration above the repository of comparable or greater magnitude than considered in DOE's calibration of infiltration uncertainty.

NRC staff compared the average reduction in precipitation that becomes net infiltration calculated for Yucca Mountain (SAR Tables 2.3.1-2 through 2.3.1-4) with reductions for other sites reported in the general literature. For Yucca Mountain, the ratio of calculated areal-average net infiltration to mean areal-average precipitation ranges from 0.022 to 0.154 for the present-day climate state, 0.023 to 0.191 for the monsoon climate state, and 0.047 to 0.166 for the glacial transition climate state. These areal-average estimates represent the entire MASSIF model domain. The NRC staff notes that these ratios are consistent with infiltration estimates from other arid and semi-arid sites in Nevada and the western United States with comparable precipitation rates [e.g., Stothoff and Musgrove (2006aa); SNL Section 7.2.1.2 (2008az)]. On the basis of the comparison of DOE estimates for Yucca Mountain with other sites in Western United States, NRC staff notes that the DOE estimates of areal-average net infiltration and its uncertainty are supported by information from relevant analog sites.

Using NRC staff's independently developed infiltration model and extensive independent staff field and modeling confirmatory investigations related to infiltration at Yucca Mountain (e.g., Stothoff, 2009aa, 2008ab, 1999aa, 1997aa, 1995aa; Stothoff, et al., 1999aa; Groeneveld, et al., 1999aa; Woolhiser, et al., 2000aa, 2006aa; Stothoff and Musgrove, 2006aa; Stothoff and Walter, 2007aa), the NRC staff further assessed DOE's calculated uncertainty regarding areal-average net infiltration estimates for the climate states used in the DOE performance assessment. The NRC staff's Infiltration Tabulator for Yucca Mountain (Stothoff, 2008aa) is an independent numerical model to estimate the uncertainty in net infiltration at Yucca Mountain. The Infiltration Tabulator for Yucca Mountain uses the same starting site characteristics of soil and bedrock maps and bedrock matrix hydraulic properties as the MASSIF model. The Infiltration Tabulator for Yucca Mountain differs from MASSIF in its computational approaches; conceptual models for water redistribution in subsurface overland flow and evapotranspiration; and input parameters for soil depths, soil and bedrock hydraulic properties, and topography.

The NRC staff compared outputs from the DOE and NRC models and notes that the models produce comparable near-lognormal distributions for calculated areal-average net infiltration when using similar uncertainty in input parameters. Similarly, both models show that soil thickness and the soil retention characteristics are the two dominant factors (after precipitation) controlling net infiltration. DOE's model and the independent NRC model have generally comparable representations for uncertainty in areal-average net infiltration.

On the basis of these comparisons, the NRC staff notes that (i) each set of 4 net infiltration maps used to represent a climate state in the first 10,000 years after closure falls within or above the range of estimates represented by other sites with comparable arid climatic conditions and (ii) DOE's uncertainty distribution reasonably represents uncertainty in areal-average net infiltration.

In support of its consideration of spatial variability in DOE's net infiltration in the vicinity of the proposed repository footprint, DOE provided an analysis in DOE Enclosure 4 (2009cr) that considers the consequences on performance from a variant property set that favors focused (channel) infiltration instead of distributed infiltration. The variant property set was based on simulations of observations in Pagany Wash. DOE considered the consequences of both spatial and temporal aspects of the variant property set with the infiltration model from two washes, stating that the consequences of a focused infiltration pattern are insignificant to repository performance calculations. The base case and variant simulations yield similar calculated values of areal-average net infiltration, with somewhat less net infiltration within the repository footprint for maps with larger total net infiltration, as described in SNL, Table 7.1.3.2-1(a) (2007az). DOE explained that the similarity in areal-average net infiltration arises from conservation of mass, in that the water infiltrating into channels in the variant case would have otherwise infiltrated into hillslopes without the enhanced runoff. In other words, redistributing water through overland flow does not appreciably increase areal-average net infiltration in DOE's model. NRC staff notes that the uncertainty in areal-average net infiltration above the repository related to different spatial patterns of infiltration is small relative to other sources of uncertainty that DOE considered. DOE showed that a reasonable alternative representation of spatial variability did not appreciably increase areal-average net infiltration.

In TER Section 2.2.1.3.5.3.2, NRC staff notes that DOE's representations of temporal variability in climate and meteorology are both reasonable for the intended use. The following discussion considers temporal variability in DOE's net infiltration estimates with respect to temporal resolution of net infiltration calculations and systematic effects on net infiltration.

The NRC staff considers the dominant way that temporal resolution may affect net infiltration estimates is in the partitioning between infiltration and runoff during precipitation events. This is because runoff calculations may be sensitive to the detailed representation of precipitation during storm events when the soil cannot accept all of the precipitation. By using a daily timestep in the net infiltration model simulations, DOE elected not to consider peak flow rates within a day, either as subsurface water movement or as runoff, as detailed in SNL Section 6.2.3 (2007az). As a consequence of not considering peak flow rates within a day, the spatial patterns of net infiltration may be affected by differences in runoff calculations. DOE supported the selected timestep by illustrating in SNL Appendix H (2007az) and DOE Enclosure 4 (2009cr) that calculated areal-average net infiltration has little sensitivity to uncertainty in the relationship between daily precipitation and precipitation duration. Because the treatment of temporal resolution of precipitation affects the partitioning between infiltration and runoff—which in turn affects the spatial patterns of infiltration—the NRC staff compared the uncertainty of different spatial patterns of infiltration with the other sources of uncertainty that DOE incorporated into the performance assessment. DOE reasonably represented temporal resolution in net infiltration calculations because (i) uncertainty in areal-average net infiltration above the repository related to different spatial patterns of infiltration is small relative to other sources of uncertainty that DOE considered and (ii) NRC staff noted in TER Section 2.2.1.3.6.3.2 that spatial variability itself does not significantly affect performance.

In summary, DOE's representation of net infiltration for the first 10,000 years is reasonable for use in performance assessment calculations. This is because the range and uncertainty of areal-average infiltration within the repository vicinity is consistent with (i) available information on infiltration in arid environments and (ii) NRC staff's independent modeling (Stothoff and Walter, 2007aa). Further, because saturated zone model results are less sensitive to uncertainty in net infiltration estimates than unsaturated zone model results, staff's evaluations regarding net infiltration results for unsaturated zone flow and transport also hold for the saturated zone. Additionally, available information suggests that DOE's representation of spatial and temporal variability in net infiltration is likely to have a small influence on release, transport, and expected dose calculations compared to areal-average infiltration.

2.2.1.3.5.4 NRC Staff Conclusions

The NRC staff notes that the DOE description of this model abstraction for climate and infiltration is consistent with the guidance in the YMRP, as supplemented by guidance in NRC (2009ab). NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.5.5 References

Allen, R.G., I.A. Walter, R.L. Elliott, T. Howell, D. Itenfisu, and M. Jensen. 2005aa. *The ASCE Standardized Reference Evapotranspiration Equation*. Reston, Virginia: American Society of Civil Engineers.

Allen, R.G., L.S. Pereira, D. Raes, and M. Smith. 1998aa. "Crop Evapotranspiration: Guidelines for Computing Crop Water Requirements." FAO Irrigation and Drainage Paper 56. Rome, Italy: Food and Agriculture Organization of the United Nations.

Bull, W.B. 1991aa. *Geomorphic Responses to Climate Change*. New York City, New York: Oxford University Press.

Christensen, J.H., B. Hewitson, A. Busuioc, A. Chen, X. Gao, I. Held, R. Jones, R.K. Kolli, W.-T. Kwon, R. Laprise, V. Magana Rueda, L. Mearns, C.G. Menendez, J. Raisanen, A. Rinke, A. Sarr, and P. Whetton. 2007aa. *In Climate Change 2007: The Physical Science Basis*. Contribution of Working Group I to the 4th Assessment Report of the Intergovernmental Panel on Climate Change. S. Solomon, D. Quin, M. Manning, Z. Chen, M. Marquis, K.B. Averyt, M. Tignor, and H.L. Miller, eds. Cambridge, United Kingdom and New York City, New York: Cambridge University Press.

DOE. 2010ai. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.5, Set 1 and (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (February 2) J.R. Williams to J.H. Sulima (NRC). ML100340034. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bo. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (June 1) J.R. Williams to J.H. Sulima (NRC). ML091530403. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cr. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.5, Set 1." Letter (June 24) J.R. Williams to J.H. Sulima (NRC). ML0918308480 and ML0918300710. Washington, DC: DOE, office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Fedors, R. 2007aa. "Soil Depths Measured at Yucca Mountain During Site Visits in 1998." Note (January 9) R. Fedors to J. Guttmann (NRC). ML063600082. Washington, DC: NRC.

Forester, R.M., J.P. Bradbury, C. Carter, A.B. Elvidge-Tuma, M.L. Hemphill, S.C. Lundstrom, S.A. Mahan, B.D. Marshall, L.A. Neymark, J.B. Paces, S.E. Sharpe, J.F. Whelan, J.F. Wigand, and P.E. Wigand. 1999aa. "The Climatic and Hydrologic History of Southern Nevada During the Late Quaternary." USGS Open-File Report 98-635. Denver, Colorado: U.S. Geological Survey.

Groeneveld, D.P., S.A. Stothoff, and R. Fedors. 1999aa. “Weedy Brome Grasses and Their Potential Effect on the Infiltration and Recharge Rates in the Vicinity of Yucca Mountain, Nevada.” San Antonio, Texas: CNWRA.

NRC. 2009ab. “Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63.” Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, “Integrated Issue Resolution Status Report.” Rev. 1. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. Washington, DC: NRC.

SNL. 2008ab. “Features, Events, and Processes for the Total System Performance Assessment: Analyses.” ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007az. “Simulation of Net Infiltration for Present-Day and Potential Future Climates.” MDL–NBS–HS–000023. Rev. 01. AD 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2006aa. “Data Analysis for Infiltration Modeling: Extracted Weather Station Data Used To Represent Present-Day and Potential Future Climate Conditions in the Vicinity of Yucca Mountain.” ANL–MGR–MD–000015. Rev. 00. ACN 01. Las Vegas, Nevada: Sandia National Laboratories.

Stothoff, S.A. 2010aa. “Infiltration and Unsaturated Zone Confirmatory Analyses.” Electronic Scientific Notebook 1005E. ML111110877. San Antonio, Texas: CNWRA.

Stothoff, S.A. 2009aa. “Subsurface Observations Related to Infiltration at Yucca Mountain.” ML093160095. San Antonio, Texas: CNWRA.

Stothoff, S.A. 2008aa. “Infiltration Tabulator for Yucca Mountain: Bases and Confirmation.” CNWRA Report 2008-001. ML082350701. San Antonio, Texas: CNWRA.

Stothoff, S.A.. 2008ab. “Compilation of Prelicensing Field Observations Related to Infiltration at Yucca Mountain and Future-Climate Analogs.” ML082000174. San Antonio, Texas: CNWRA.

Stothoff, S.A. 1999aa. “Infiltration Abstractions for Shallow Soil Over Fractured Bedrock in a Semiarid Climate.” ML061370364. San Antonio, Texas: CNWRA.

Stothoff, S.A. 1997aa. “Sensitivity of Long-Term Bare Soil Infiltration Simulations to Hydraulic Properties in an Arid Environment.” *Water Resources Research*. Vol. 22, No. 4. pp. 547–558.

Stothoff, S.A. 1995aa. NUREG/CR–6333, “BREATH Version 1.1—Coupled Flow and Energy Transport in Porous Media: Simulator Description and User Guide.” Washington, DC: NRC.

Stothoff S. and M. Musgrove. 2006aa. “Literature Review and Analysis: Climate and Infiltration.” CNWRA Report 2007-002. ML063190115. San Antonio, Texas: CNWRA.

Stothoff, S. and G. Walter. 2007aa. "Long-Term-Average Infiltration at Yucca Mountain, Nevada: Million-Year Estimates." CNWRA Report 2007-003. ML072760607. San Antonio, Texas: CNWRA.

Stothoff, S.A., D. Or, D.P. Groeneveld, and S.B. Jones. 1999aa. "The Effect of Vegetation on Infiltration in Shallow Soils Underlain by Fissured Bedrock." *Journal of Hydrology*. Vol. 218. pp. 169–190.

Winograd, I.J., A.C. Riggs, and T.B. Coplen. 1998aa. "The Relative Contributions of Summer and Cool-Season Precipitation to Groundwater Recharge, Spring Mountains, Nevada, USA." *Hydrogeologic Journal*. Vol. 6, No. 1. pp 77–93.

Woolhiser, D. R.W. Fedors, R.E. Smith, and S.A. Stothoff. 2006aa. "Estimating Infiltration in the Upper Split Wash Watershed, Yucca Mountain, Nevada." *Journal of Hydrologic Engineering.* Vol. 11, No. 2. pp 123–133.

Woolhiser, D.A., S.A. Stothoff, and G.W. Wittmeyer. 2000aa. "Estimating Channel Infiltration From Surface Runoff in the Solitario Canyon Watershed, Yucca Mountain, Nevada." *Journal of Hydrologic Engineering*. Vol. 5, No. 3. pp. 240–249.

CHAPTER 9

2.2.1.3.6 Unsaturated Zone Flow

2.2.1.3.6.1 Introduction

This chapter of the Technical Evaluation Report (TER) addresses the U.S. Department of Energy's (DOE's) abstraction of groundwater flow in that portion of the repository system above the water table. DOE presented this information in its Safety Analysis Report (SAR) of June 3, 2008 (DOE, 2008ab), and subsequent update of February 19, 2009 (DOE, 2009av). Although information from other sections of the SAR is cited in the review of the unsaturated zone flow abstractions, the primary SAR Sections used are 2.3.2 (Unsaturated Zone Flow), 2.3.3 (Water Seeping into Drifts), and 2.3.5.4 (In-Drift Thermohydrological Environment).

The proposed Yucca Mountain repository site has up to 400 m [1,300 ft] of variably saturated rock between the ground surface and the repository, and at least 200 m [650 ft] between the repository and the underlying water table (SAR Sections 2.1.1.1 and 2.1.1.3). Water percolating through the unsaturated zone may enter the drifts, thereby providing the means to interact with and potentially corrode the waste packages. Water percolating through the unsaturated zone below the repository also provides flow pathways for transporting radionuclides downward to the water table. Once radionuclides pass below the water table, they may move laterally within the saturated zone to the accessible environment. In this chapter, the term "unsaturated zone flow" includes not only flow processes in the host rock under ambient and thermally perturbed conditions, but also in-drift hydrological processes related to flow through natural rubble and in-drift convection and condensation. Unsaturated flow both above and below the repository horizon is addressed in this chapter.

The unsaturated zone plays a role in two of the DOE-defined barriers: the Upper Natural Barrier and the Lower Natural Barrier (SAR Section 2.3.2). These barriers are reviewed in TER Sections 2.2.1.1.3.2. Together with Climate and Infiltration (reviewed in TER Section 2.2.1.3.5), processes in the unsaturated zone above the repository comprise the Upper Natural Barrier. They influence system performance through the amount of water reaching the Engineered Barrier System and their control on hydrological conditions in the drift. In DOE's model of the nominal scenario, the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms (reviewed in TER Section 2.2.1.3.3), Degradation of Engineered Barriers (reviewed in TER Section 2.2.1.3.1), and Radionuclide Release Rates and Solubility Limits (reviewed in TER Section 2.2.1.3.4) abstractions use in-drift liquid water, relative humidity, and temperature to assess the potential for corrosion of waste packages, release of waste, and transport to the natural system. In the disruptive scenarios of seismic and igneous intrusion (reviewed in TER Sections 2.2.1.3.2 and 2.2.1.3.10), DOE's model uses the flux of water to assess the movement of radionuclides to the natural system below the repository. The portion of the unsaturated zone below the repository is part of the Lower Natural Barrier. The magnitude and distribution of flux in the unsaturated zone below the repository are used to determine the flow pathways for Radionuclide Transport in the Unsaturated Zone (reviewed in TER Section 2.2.1.3.7). The unsaturated zone below the repository links the repository Engineered Barrier System to the Saturated Zone Flow and Transport System (reviewed in TER Sections 2.2.1.3.8 and 2.2.1.3.9) and ultimately to the biosphere in the accessible environment (reviewed in TER Sections 2.2.1.3.12 to 2.2.1.3.14).

The purpose of this chapter is to review DOE estimates of the

- Magnitude and distribution of the mass flux of water (percolation) moving through the unsaturated zone and reaching the drift

- Amount and distribution of liquid water seeping into the drift, contacting the engineered barriers (i.e., drip shield), and becoming available to carry radionuclides out of the drift and into the natural environment

- Environmental conditions inside the drift (i.e., temperature, relative humidity, and moisture redistribution and condensation)

- Magnitude and distribution of flux in the unsaturated zone below the repository as is important for transport of radionuclides

2.2.1.3.6.2 Evaluation Criteria

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for unsaturated zone flow, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments).

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Section 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

For this model abstraction of unsaturated zone flow, 10 CFR 63.342(c)(1) provides additional requirements for assessing the effects of seismic and igneous activity on the repository performance, subject to the probability limits in 10 CFR 63.342(a) and (b). Specific constraints on the seismic and igneous activity analyses are in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

In addition, for this model abstraction of unsaturated zone flow, 10 CFR 63.342(c)(2) addresses the assessment of climate change after 10,000 years by using a constant-in-time specification of the mean and uncertainty distribution for repository-average deep percolation rate for the period from 10,000 to 1 million years. DOE elected to use this representation in its SAR. Thus, implementation of the specified representative percolation rate and its uncertainty distribution is reviewed for the post-10,000-year period.

The NRC staff's review of the SAR and supporting information follows the guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Section 2.2.1.3.6, Unsaturated Zone Flow, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of unsaturated zone flow are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff's review used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of the unsaturated zone flow important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as assessed by the NRC staff, are discussed in detail in this chapter. The NRC staff's assessment is based both on risk information provided by DOE and on NRC staff knowledge gained through experience and independent analyses.

2.2.1.3.6.3 Technical Evaluation

The review of the technical information DOE provided for unsaturated zone, seepage, and in-drift hydrological conditions in this section is divided into six subsections. The first subsection is an overview of the DOE description of processes and models, and a summary of results for the entire unsaturated zone flow area of review. The overview (TER Section 2.2.1.3.6.3.1) provides context for and reviews the integration between models and results separately evaluated in the remaining five subsections (TER Sections 2.2.1.3.6.3.2–2.2.1.3.6.6) within the unsaturated zone. The remaining five subsections follow a natural flow through the unsaturated zone system: (i) ambient flow above the repository, (ii) thermohydrology, (iii) ambient and thermal seepage, (iv) in-drift hydrologic conditions, and (v) ambient flow below the repository.

2.2.1.3.6.3.1 Integration Within the Unsaturated Zone

In this section, the NRC staff's review covers a range of processes and features occurring at widely disparate spatial and temporal scales within the Upper and Lower Natural Barriers and, to a lesser extent, within the Engineered Barrier System. Within this area of review, the NRC staff evaluates aspects of repository performance pertaining to

- Unsaturated zone flow fields (SAR Section 2.3.2) above and below the repository

- Seepage into drifts (SAR Section 2.3.3)

- Hydrological aspects of the in-drift environment (SAR Section 2.3.5); in particular, the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4.1), the In-Drift Condensation Model (SAR Section 2.3.5.4.2), and the thermohydrologic response to the range of design thermal loadings (SAR Section 2.3.5.4.3)

The NRC staff evaluates the DOE treatment of interactions between liquid fluxes and engineered barriers inside drifts (i.e., drip shields, waste packages, and inverts) in TER Sections 2.2.1.3.3 and 2.2.1.3.4.

DOE's unsaturated zone flow models receive input from and provide output to several areas of review. SAR Section 2.3.1 provided spatially distributed net infiltration rates for the different predicted future climates for use as the top boundary flux of models in the unsaturated zone. For outputs, in-drift liquid-phase water, relative humidity, and temperature were used for abstractions of (i) chemistry for the in-drift environment (SAR Sections 2.3.5.3, 2.3.5.5), (ii) corrosion of engineered components (SAR Section 2.3.6), and (iii) waste form degradation and in-drift transport (SAR Section 2.3.7) in the Total System Performance Assessment (TSPA). In SAR Section 2.3.5, feedback from thermal-hydrological-chemical models in the host rock during the thermal period provided information on the perturbation of hydrological properties caused by emplaced waste. Output flow fields from the ambient unsaturated zone mountain-scale model (SAR Section 2.3.2), along with radionuclide flux from the Engineered Barrier System (SAR Section 2.3.7), were then used by Radionuclide Transport in the Unsaturated Zone abstraction (SAR Section 2.3.8).

This TER section focuses on higher level issues common to each of the unsaturated zone flow models, including (i) integration among those models, (ii) representative flow reduction through the mountain, and (iii) propagation of uncertainty in performance assessment calculations.

Integration of Unsaturated Zone Flow Models

DOE represented water and heat transfer in the unsaturated zone using a variety of process-level models covering features and processes at a range of scales from millimeters to kilometers [fraction of inches to miles]. In addition to models that provide direct input, DOE used additional models to support aspects of conceptual model assumptions and parameter input. The models require different computational grids, different data needs, and different model support. Because the different models overlap in terms of features, processes, inputs, and outputs, the NRC staff reviewed the DOE integration between models. Specifically, the

NRC staff evaluated the spatial continuity of percolation flux (consistent propagation of high and low flux patterns) and quantification of barrier capability through the mountain.

The following list of models, inputs, and outputs used in performance assessment calculations provides context to the parts of TER Section 2.2.1.3.6.3. NRC staff evaluation in the remainder of TER Section 2.2.1.3.6.3 parallels this list, which follows the flow of water through the mountain.

Ambient Site-Scale Unsaturated Flow Model (SAR Section 2.3.2)

- Receives top flux boundary condition from net infiltration model
- Creates set of flow fields for TSPA
    - Above-repository flux distribution to the Multiscale Thermal-Hydrologic Model
    - Below-repository flow field for unsaturated zone transport (see last item in this list)
- Reviewed by NRC staff in TER Sections 2.2.1.3.6.3.2 (Above Repository Only)

Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4)

- Composed of a set of five linked process-level thermal and thermohydrology models
- Creates a set of thermal response abstractions that provides
    - In-drift temperature and relative humidity for chemistry of seepage and corrosion of engineered barriers abstractions
    - Deep percolation field for seepage and chemistry models
    - Flux from host rock into invert for Engineered Barrier System transport
- Reviewed by NRC staff in TER Section 2.2.1.3.6.3.3

Ambient and Thermal Seepage Models (SAR Section 2.3.3)

- Process-level models used to create seepage abstractions for TSPA that provide
    - Seepage fraction (number of waste packages getting wet) to Engineered Barrier System transport
    - Seepage flux to Engineered Barrier System transport
    - Temperature threshold at drift wall, above which no seepage occurs
- Reviewed by NRC staff in TER Section 2.2.1.3.6.3.4

In-Drift Convection and Condensation Models (SAR Section 2.3.5.4)

- Convection model provides dispersion coefficients to condensation model

- Condensation model provides flux rate and distribution to Engineered Barrier System transport

- Reviewed by NRC staff in TER Section 2.2.1.3.6.3.5

Site-Scale Unsaturated Flow Below the Repository (see first item in this list)

- Provides flow field to unsaturated zone transport model
- Reviewed by NRC staff in TER Section 2.2.1.3.6.3.6

The NRC staff reviews repository performance with respect to water and heat transfer within the unsaturated zone by separately evaluating the individual process models and their abstractions (TER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6) and the overall integration among models (the present section). The NRC staff recognizes that it may not be practical to use one process-level model to represent the entire suite of features and processes in the unsaturated zone for performance assessment calculations. The NRC staff also recognizes that linking decoupled models is a standard practice for modeling complex systems at a wide expanse of scales, when important dependencies from upstream to downstream models are adequately reflected in linked models. On the basis of staff knowledge of general modeling practices and staff review of the information that links (is passed between) separate DOE models in TER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6, the NRC staff notes that the ensemble of DOE models is a reasonable approach for representing the unsaturated system for the performance assessment.

NRC staff reviewed the information passed between the unsaturated zone models. DOE did not strictly follow continuity between models that relate to the transfer of water from the natural system into the emplacement drifts and back to the natural system. However, spatial continuity was approximately followed between DOE's larger site-scale models and smaller drift-scale seepage models. DOE used five percolation bins to maintain continuity of flow above, through, and below drifts for both ambient and thermal periods. Spatial detail is lost in the progression from the net infiltration model to the site-scale models and the seepage model, but use of the percolation bin approach ensures that high and low seepage zones correspond generally with high and low percolation zones above and below drifts. The NRC staff evaluations of spatial variability, upscaling, downscaling, and other linkages between models, as appropriate, are included in TER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.4. In those TER sections, NRC staff notes that DOE's treatment of spatial variability is reasonable for performance assessment because the inclusion of more detailed variation did not significantly increase calculated dose. On the basis of the NRC staff review of the DOE technical bases, the NRC staff notes that the flux of water is adequately integrated between the unsaturated zone models and between the natural system and the Engineered Barrier System.

NRC staff review also considered the DOE implementation of barrier capabilities, represented by changes to percolation flux rates as water moves through the mountain from the ground surface into the drifts and onto the water table. Table 9-1 illustrates the quantitative reduction in flux from the ground surface to water entering the drift using flux averages over the repository footprint. Flux values in the table are from DOE Enclosure 1, Tables 1, 5, and 8 (2010ai), and seepage fraction values are from SAR Tables 2.1-6 to 2.1-9. The table also provides the component of the Upper Natural Barrier, primary features or processes, and the relevant SAR section for each step of the flux reduction.

DOE presented seepage flux values as volumetric flux over the area of a waste package, calculated by multiplying seepage flux values determined in units of volumetric flux per unit area by a waste package footprint that is 5.1 m [17 ft] long and 5.5 m [18 ft] wide (i.e., drift width).

To maintain consistent units of flux for this table, the NRC staff divided the DOE-provided seepage flux values by the same scaling factor. DOE Enclosure 1, Tables 5 and 8 (2010ai) used million-year simulation results to provide the seepage flux and fraction values for the 10,000-year period, the latter of which may differ slightly from the simulation results used in TSPA calculations for 10,000-year dose estimates. Flux values of net infiltration through deep percolation retain the significant figures DOE presented. In Table 9-1, seepage fraction is that portion of the drifts where dripping is predicted to occur (also called the seeping environment).

The table includes DOE values for its nominal case (no disruptive events) and seismic ground motion scenario. In its igneous intrusion scenario (not included in the table), DOE conservatively assumed that seepage processes at the drift wall do not act as a barrier. Condensation rates are similarly not included in Table 9-1, because these fluxes are short lived. In DOE’s model these have no effect on performance, because drip shields are predicted to remain intact beyond the several thousand years to 10,000 years of the thermal period. The

Table 9-1. Quantitative Reduction in Flux From the Ground Surface to Water Entering the Drift Using Flux Averages Over the Repository Footprint

	Precipitation mm/yr*	Net Infiltration mm/yr*	Unsaturated Zone Site-Scale Top Boundary Net Infiltration mm/yr*	Deep Percolation mm/yr*	Seepage Repository Footprint	
					Flux mm/yr*	Fraction of Area
Component of Upper Natural Barrier	—	Topography and Soils	—	Unsaturated Zone	Unsaturated Zone	
Primary Feature or Processes	Semiarid Climate	Evapo-transpiration, Runoff, Infiltration	Uncertainty in Net Infiltration	—	Capillary Diversion and Vapor Barrier	
Section of SAR	2.3.1	2.3.1	2.3.2	2.3.2	2.3.3	
Thermal Period†	—	—	—	—	0	0
At 10,000 years‡ Nominal	296.7	38.88	21.37	21.74	2.0 {6.4}§	0.31
At 10,000 years‡ Seismic					2.3 {7.4}§	0.31
Post-10,000 years, Nominal	—	—	—	31.83	3.4 {8.5}§	0.40
Post-10,000 years, Seismic					15 {22}§	0.69

*Units: 25.4 mm/yr = 1 in/yr
†Thermal period defined by drift wall temperature > 100 °C [212 °F] (SAR Section 2.3.3.3.4).
‡Values of precipitation and percolation for initial 10,000 years are for glacial transition climate.
§Average flux for seeping environment is in brackets.

NRC staff noted in TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that DOE reasonably supported that drip shields are expected to remain intact for at least 12,000 years and thus will be able to divert all water away from waste packages. Representative thermal effects (peak waste package temperature, drift wall temperature, duration of drift wall temperatures at or above boiling) caused by waste package emplacement were summarized in SAR Tables 2.3.5-7 and 2.3.5-8 and are not reflected in Table 9-1.

NRC staff used the list of average flux values in Table 9-1 as a first-order continuity check for net infiltration, deep percolation, and seepage in DOE's process models. The first-order continuity check the NRC staff considered in its evaluation is an evaluation of the magnitude of change in flux as new processes and features are introduced in the DOE models for the unsaturated zone. NRC staff used DOE's glacial transition climate to represent the first 10,000 years because (i) the monsoon and glacial transition climates have approximately equal flux rates and DOE's climate model represents these periods as encompassing most (94 percent) of the first 10,000 years of performance and (ii) DOE described seepage as precluded by above-boiling conditions at the repository horizon during the preceding climate state. From the entries in Table 9-1, NRC staff calculates that deep percolation is 7 percent of precipitation on average for the glacial transition climate in DOE's performance assessment model. This reduction comes from the two process-model steps that DOE used in its performance assessment: (i) net infiltration calculations, using the Mass Accounting System for Soil Infiltration and Flow (MASSIF) infiltration model, and (ii) net infiltration uncertainty calibration, using the unsaturated zone model top boundary net infiltration.

For the first 10,000 years of performance

- The net infiltration calculated by DOE for the first 10,000 years is a small fraction (approximately 13 percent) of total precipitation. The NRC staff notes in TER Section 2.2.1.3.5 that the net infiltration estimates are adequately supported because the estimates are generally consistent with infiltration for areas throughout the desert Southwest of the United States with similar climates, and with the NRC staff independent analyses conducted over the past 15 years.

- DOE compared subsurface observations with simulations based on the site-scale ambient unsaturated flow model to calibrate the uncertainty in net infiltration, leading to a further reduction of 45 percent in the percolation flux. This further reduction of net infiltration flux is reasonable because DOE showed that the reduction did not lead to an underestimate in dose (see TER Section 2.2.1.3.6.3.2).

  The average seepage flux calculated by DOE is approximately 10 percent of average percolation. Using this value, DOE calculated that 31 percent of waste package locations would become wet. NRC staff notes in TER Section 2.2.1.3.6.4 that an underestimate of seepage has a low consequence for repository performance, because DOE predicts that the drip shields remain intact and prevent seepage water from contacting the waste packages well beyond 10,000 years.

For the post-10,000-year period, NRC staff's review includes the seismic ground motion and igneous intrusion modeling scenarios, because these two scenarios contribute most to DOE's predicted dose. In particular

- Deep percolation values used for the post-10,000-year period adequately reflect the prescribed values (see TER Section 2.2.1.3.6.3.2)

- The average seepage fraction for the seismic ground motion modeling case (69 percent) is sufficiently large that increases are not expected to significantly increase expected dose (see TER Section 2.2.1.3.6.3.4)

- The average seepage flux estimated for the seismic ground motion modeling case is sufficiently large (47 percent of percolation) such that larger values would not significantly affect expected dose (see TER Section 2.2.1.3.6.3.4)

- DOE represented emplacement drifts as not forming a barrier with respect to seepage for the igneous intrusion modeling case; therefore, seepage is not underestimated (see TER Section 2.2.1.3.6.3.4)

On the basis of these considerations, the NRC staff notes that the average values of flux reduction are consistent with staff understanding of the features, processes, barrier capabilities of the unsaturated zone, and sensitivity to repository performance.

In summary, the NRC staff notes that at a high level, flux variability is adequately incorporated throughout the unsaturated zone flow models discussed previously. Flux variability was evaluated both in terms of spatial continuity of high and low flux zones and in terms of reductions in average flux through the mountain. Because continuity of flux through the mountain and average reductions in flux are reasonable, the DOE collection of linked process models and abstractions for site-scale unsaturated flow and seepage is reasonable for its intended use in performance assessment. Detailed NRC staff review of repository performance with respect to DOE results for net infiltration, percolation, seepage, and condensation is given in TER Sections 2.2.1.3.5.3 and 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6.

Propagation of Uncertainty in Performance Assessment

NRC staff evaluates propagation of uncertainty in DOE performance assessment calculations by examining (i) the technical basis for selecting parameters and uncertainties for sensitivity analyses and (ii) the DOE sensitivity analysis for the selected parameters with respect to TSPA intermediate results and calculated doses.

DOE conducted a series of analyses to determine the sensitivity of model outputs to input parameter uncertainties in its performance assessment calculations (SAR Section 2.4.2.3.3.3). DOE treated a subset of parameters in the unsaturated zone and Engineered Barrier System as uncertain in the analyses. These parameters relate to infiltration (linked to percolation), seepage into the drift, host rock thermal and hydrogeologic parameters, and in-drift thermal processes that affect the release of radionuclides. In TER Sections 2.2.1.3.5.3.3 and 2.2.1.3.6.3.2–2.2.1.3.6.3.6, the NRC staff evaluated (i) the analyses used to identify uncertain parameters carried into performance assessment calculations and (ii) the technical basis for describing uncertain parameters when evaluating individual models and abstractions. In each of these sections, the NRC staff noted that, for use in the corresponding abstraction, DOE addressed model and parameter uncertainty in performance assessment calculations. On the basis of staff's analyses of NRC Appendix D (2005aa), NRC staff notes that DOE identified a reasonable set of the most important parameters to include in performance assessment calculations of dose. Because the uncertainty for the individual models is reasonable, DOE adequately propagated uncertainty throughout the unsaturated zone.

Further confidence that DOE reasonably propagated uncertainty in its estimation of dose is based on the DOE estimate of the importance of the unsaturated zone compared to Engineered Barrier System components. DOE evaluated the sensitivity of intermediate results and expected mean annual dose as calculated by TSPA with respect to the selected parameters. In general, the DOE analyses identified key uncertain input parameters associated with waste package failure as the dominant factors affecting performance (SAR Table 2.4-12); saturated zone transport and net infiltration were also identified as key uncertain input parameters under some scenarios.

The DOE analyses consistently identified uncertainty in net infiltration, which is closely related to percolation fluxes within the unsaturated zone, as significantly affecting uncertainty in intermediate results (e.g., drift seepage, drift wall temperatures, radionuclide releases from the Engineered Barrier System, unsaturated zone radionuclide transport rates) and expected mean annual doses (SNL, 2008ag). The DOE analyses also identified (i) relatively smaller contributions to uncertainty in seepage into drifts arising from uncertainty in host rock permeability and capillary strength and (ii) contributions to uncertainty in in-drift temperature and relative humidity from host rock thermal conductivity, as outlined in SNL Section K4 (2008ag).

The NRC staff examined the DOE sensitivity analyses with respect to intermediate results and expected mean annual dose by comparing the sensitivity results with the DOE description of the physical processes governing barrier function as represented in the models used for performance assessment. The NRC staff's review focused on parameters that DOE identified as systematically affecting either intermediate results or expected mean doses. On the basis of the NRC staff comparison, the NRC staff notes that the DOE sensitivity analysis is consistent with the DOE description of the physical processes embodied in models used for performance assessment calculations with respect to net infiltration, percolation fluxes, thermal responses in the host rock, seepage, and in-drift temperature and humidity.

The NRC staff's review of the DOE procedures for propagating uncertainty in performance assessment considers the reasonableness of overall performance assessment results. This is based in part on the NRC staff review of the performance assessment calculations (TER Section 2.2.1.4.1). DOE's rankings of key parameter inputs are consistent with its representation of engineered barrier characteristics and are derived from a variety of approaches for sensitivity analyses, as described in SNL Appendix K9 (2008ag). The rankings suggest that failure and release mechanisms are more important to repository performance than natural system factors of the unsaturated zone, because of the longevity of the engineered barriers. NRC staff gained confidence in the reasonableness of DOE rankings of key parameter inputs on the basis of a comparison with NRC staff's parameter rankings derived from uncertainty analyses performed using independent methods and models (NRC, 2005aa).

On the basis of these considerations, DOE has adequately propagated data uncertainty associated with infiltration; seepage into the drift; host rock thermal and hydrogeologic parameters; and in-drift thermal, chemical, and mechanical processes that affect the release of radionuclides in its performance assessment.

2.2.1.3.6.3.2 Ambient Mountain-Scale Flow Above the Repository

DOE represented the unsaturated zone and Engineered Barrier System using a hierarchy of far-field, near-field, and in-drift models. DOE uses the term far-field models to focus on features and processes of the natural system sufficiently distant from the repository to be unaffected by excavation and emplacement of waste. Similarly, near-field models focus on features and processes in the region affected by excavation and emplacement of waste (these terms are further described in the glossary). In-drift models focus on features and processes inside the disposal drift. DOE used its site-scale unsaturated zone flow model to represent far-field ambient mountain-scale flow from the ground surface to the water table (i.e., above, within, and below the proposed repository). In DOE documentation, site-scale and mountain-scale flow are interchangeable terms referring to the large scale of the computational grid, and ambient flow is the percolation flux that occurs without the flow-diverting effects of drifts and waste-produced thermal boiling fronts. DOE used output from the site-scale unsaturated zone flow model to account for far-field effects. This TER section evaluates repository performance with respect to ambient water flow within the upper unsaturated zone between the ground surface and the proposed repository horizon, focusing on the site-scale unsaturated zone flow model and an intermediate-scale model that links mountain-scale flow to seepage models. The NRC staff evaluates repository performance related to the effects of drift openings and thermal perturbation on flow patterns within and below the proposed repository horizon in TER Sections 2.2.1.3.6.3.3 through 2.2.1.3.6.3.6.

In its review, NRC staff considered the site-scale unsaturated-zone flow model from two perspectives: (i) in the context of flow within the unsaturated zone as a whole and (ii) in the context of the DOE-defined upper and lower natural barriers. The NRC staff evaluation in this TER section focuses on aspects of repository performance primarily related to the upper natural barrier, in particular aspects of flow that affect seepage. The evaluation presented in TER Section 2.2.1.3.6.3.6 addresses aspects of repository performance primarily related to the lower natural barrier; in particular, aspects of flow that affect transport. The present section also considers some aspects of repository performance that apply to the entire unsaturated zone, such as estimation of parameter value sets for models. This TER section also describes the NRC staff evaluation of aspects of the repository performance specifically related to DOE's implementation of the prescribed percolation values during the period from 10,000 to 1 million years after permanent closure.

Conceptual Model

The DOE conceptual model for flow in the unsaturated zone is based on the primary hydrogeologic units within a column (SAR Section 2.3.2.2.1). The uppermost, the Tiva Canyon welded unit, features vertical episodic and fracture-dominated flow strongly influenced by episodic infiltration pulses. The underlying Paintbrush Tuff nonwelded (PTn) unit exhibits essentially vertical matrix-dominated flow with a strong potential for dampening and smoothing flows, thereby buffering lower units from episodic and localized infiltration pulses. The repository host horizon is within the Topopah Spring welded (TSw) unit, which features essentially vertical fracture-dominated flows in equilibrium with decadal-average net infiltration. In the TSw, mountain-scale flow patterns are controlled by mountain-scale infiltration patterns and fine-scale flow patterns are controlled by the TSw rock properties. The DOE conceptual model for the units underlying the repository host horizon [Calico Hills nonwelded (CHn) and Crater Flat undifferentiated (CFu) units] is discussed and evaluated in detail in TER Section 2.2.1.3.6.3.6.

DOE adequately described its conceptual model of the physical phenomena affecting unsaturated flow under ambient conditions because DOE described (i) the features of the site geology that affect unsaturated zone flow; (ii) the most important processes affecting the performance of the upper natural barrier, such as lateral diversion, spatial variability, and temporal variability; and (iii) the body of laboratory and field data supporting the conceptual model. The NRC staff gained additional confidence that DOE's description of its conceptual model of the physical phenomena was reasonable by comparing the DOE description of the ambient-flow conceptual model with the NRC staff understanding of the Yucca Mountain natural system, obtained from extensive prelicensing experience and independent analyses of unsaturated zone flow processes at Yucca Mountain (e.g., NRC, 2005aa).

Implementation of the Conceptual Model

DOE used several models to represent various aspects of flow within the upper unsaturated zone, each considering different processes and scales. Because several DOE abstractions consider flows in the upper unsaturated zone, and multiple downstream models depend on the calculated flows, the NRC staff considered how the flow abstractions interact with the other calculations affecting repository performance by (i) separately evaluating individual flow abstractions, (ii) evaluating downstream uses of the flow abstractions for consistency between abstractions, and (iii) evaluating the effects of the abstractions on performance assessment calculations as a whole. In this TER section, the NRC staff evaluates the ambient mountain-scale flow model, focusing on the linkages between the model and both upstream models (i.e., the infiltration model) and downstream models (e.g., the Multiscale Thermal-Hydrologic Model and seepage models).

In performing a risk-informed, performance-based review of DOE's representation of ambient flow in the upper unsaturated zone, the NRC staff considered the DOE-defined function of the Upper Natural Barrier to prevent or substantially reduce water seeping into emplacement drifts. On the basis of the function of the barrier, the NRC staff identified several aspects of the DOE abstractions as having primary importance for representing flow in the upper unsaturated zone. These aspects include

- Integration of flow models; in particular, integration between the mountain and seepage scales, because integration directly affects modeling approaches in upstream and downstream models

- Mountain-scale flow patterns within the upper unsaturated zone, because this characteristic affects the amount of water diverting away from the repository as a whole

- Site-scale unsaturated zone flow model parameters, because model properties may affect flow patterns and thereby affect the rate of water seepage into drifts or the thermal response of the natural system

- Systematic effects of infiltration uncertainty on percolation fluxes, because percolation rates at the repository horizon directly affect the rate of water seepage into drifts, and percolation rates below drifts directly affect the rate of radionuclide transport from the Engineered Barrier System to the water table

- Temporal and spatial variability; in particular, flow patterns at the drift walls, because these flow characteristics affect the rate and patterns of water seepage into drifts

The following sections describe the NRC staff evaluation of these aspects of repository performance with respect to ambient flow in the upper unsaturated zone.

Integration of Flow Models

The NRC staff considered the DOE approach to integrating the site-scale unsaturated zone flow model representation of ambient flow above the repository with (i) the MASSIF infiltration model (SAR Section 2.3.1), (ii) the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4), (iii) seepage models, and (iv) ambient flow below the repository. The NRC staff identified the integration of ambient flow above the repository with seepage models as the most risk significant among these for the following reasons:

- The MASSIF net infiltration model, site-scale unsaturated zone flow model, and Multiscale Thermal-Hydrologic Model are decoupled models that are linked in only one direction (see TER Section 2.2.1.3.6.3.1); the site-scale unsaturated zone flow model seamlessly considers ambient flow above and below the repository, and the linkages locally conserve mass.

- Performance assessment calculations are sensitive to seepage, which is in turn sensitive to the representation of fluxes near emplacement drifts at scales smaller than represented by the site-scale unsaturated zone flow model.

The NRC staff notes that DOE adequately integrated the site-scale unsaturated zone flow model representation of ambient flow above the repository with the MASSIF infiltration model and the Multiscale Thermal-Hydrologic Model and ambient flow below the repository because the linkages locally conserve mass and the overlapping input properties are generally consistent between the models. The NRC staff reviewed input properties that are not consistent (e.g., hydraulic properties of bedrock near the ground surface) between the infiltration and site-scale ambient unsaturated flow models. On the basis of NRC staff expertise with developing input property sets for the different needs of infiltration and unsaturated zone models (Stothoff and Walter, 2007aa; Manepally, et al., 2004aa) and knowledge of scaling of properties between different spatial scales, the NRC staff notes the differences are not important or are technically justified.

In considering integration between models, the NRC staff identified a potential concern with integration between the infiltration and seepage models (even though the two models are not directly linked), because (i) the MASSIF calculations are performed at a finer spatial and temporal resolution than the site-scale unsaturated zone flow model, (ii) modeled spatial and temporal variability in flow is reduced in the transfer of information between the two models, and (iii) spatial and temporal variability in flow near emplacement drifts affects the DOE seepage calculations.

The NRC staff evaluates DOE's integration between flow at the mountain and seepage scales, also considering integration between infiltration and seepage, in the remainder of this subsection. The NRC staff evaluates the DOE approach to modeling seepage in TER Section 2.2.1.3.6.3.4.

DOE used a hierarchical approach to link percolation fluxes within the repository horizon at the mountain scale to seepage, progressing from mountain-scale flow to the smaller scales of intermediate- or drift-scale flow and fine-scale flow (SAR Section 2.3.3.2). The site-scale unsaturated zone flow model, which calculates mountain-scale flow, represents flow averaged over an area approximately corresponding to the combined area of a drift and the pillar between drifts. Intermediate-scale flow represents average flow over an area approximately corresponding to the width of a drift. Fine-scale flow represents flow at scales smaller than a drift wall. DOE considered all three scales in calculating average seepage into drifts.

DOE represented both intermediate-scale and fine-scale flow statistically, as described in SAR Section 2.3.3.2.3 and DOE Enclosure 4 (2009bo). DOE derived an abstraction for the statistical distribution of intermediate-scale fluxes from numerical model calculations considering hydraulic-property heterogeneity in the densely welded TSw unit above the proposed repository using model parameters based on field observations. DOE provided simulation results demonstrating that the statistical patterns of predominantly vertical unsaturated zone flow at the intermediate scale stabilize within a short vertical distance from the top boundary and geological unit changes, using several representations of the top boundary flux (SAR Section 2.3.3.2.3.5). DOE also performed sensitivity studies considering alternative statistical representations of intermediate-scale variability. The NRC staff evaluates the sensitivity studies in a subsequent subsection (Temporal and Spatial Variability), noting that DOE showed that the dose consequences from alternative statistical representations of intermediate-scale variability (including potential links between infiltration and seepage) are not significant to performance.

The NRC staff notes that DOE adequately integrated mountain-scale flow and seepage for the following reasons:

- The procedure DOE used to pass flow from the mountain scale to the intermediate scale to the fine scale does not propagate flow diversion near the drift wall to larger scales, so seepage calculations use upper-bound estimates for local percolation flux.

- In performance assessment calculations, DOE used statistical distributions of relevant hydrological input parameters at a smaller scale that conserve fluxes calculated at the larger scale (e.g., mountain-scale flow model to intermediate-scale model, intermediate-scale model to seepage model).

- DOE used site observations to derive the statistical distributions for rock properties used as input to the intermediate-scale model linking the mountain-scale and seepage models.

- The performance assessment calculations are not strongly sensitive to the representation for intermediate-scale fluxes. DOE showed that, for a given areal-average seepage flux, calculated peak cumulative areal-average radionuclide releases would not be underestimated using the performance assessment procedure, outlined in DOE Enclosures 2 and 4 (2009bo). Further, DOE showed that alternative statistical representations for the intermediate-scale flux linking mountain-scale and seepage fluxes do not significantly affect the performance assessment (see the subsection Temporal and Spatial Variability in the Upper Unsaturated Zone in this TER section).

Mountain-Scale/Site-Scale Flow Patterns

The site-scale unsaturated zone flow model represents mountain-scale flow as steady state in equilibrium with climatic conditions, with water flowing from the upper boundary to the water table without lateral inflow or outflow. The NRC staff identified the key uncertainties in mountain-scale flow as (i) the magnitude of areal-average percolation flux through the repository footprint and (ii) the spatial patterns of percolation flux through the repository footprint. The NRC staff evaluates the site-scale model separately for the Upper Natural Barrier portion of the unsaturated zone, above the proposed repository (this section), and the Lower Natural Barrier portion of the unsaturated zone, below the repository (TER Section 2.2.3.6.3.6). The NRC staff evaluates the DOE representation for the spatial patterns of percolation flux in this section and evaluates the DOE representation for the magnitude of areal-average percolation flux through the repository footprint, which is determined by net infiltration, in a subsequent subsection (Infiltration Uncertainty).

DOE represents spatial variability in percolation fluxes as dominated by intermediate- and fine-scale variability in the seepage calculations for performance assessment. In the preceding section of this chapter (Integration of Flow Models), the NRC staff notes that DOE integrated mountain-scale flow and seepage. DOE accounts for alternative intermediate- and fine-scale spatial patterns of percolation fluxes within the repository footprint (and as a logical consequence acceptably accounting for alternative mountain-scale spatial patterns that preserve the areal-average percolation within the repository footprint). Therefore, the NRC staff's review of the site-scale unsaturated zone flow model representation of mountain-scale flow patterns focuses on DOE's representation of potential systematic mountain-scale flow diversion above the proposed repository horizon.

DOE showed that the site-scale unsaturated zone flow model calculates mountain-scale spatial patterns of percolation fluxes, combining matrix and fracture flows, that are essentially vertical from the ground surface through the repository horizon within the repository footprint, in accordance with the DOE conceptual model, as shown in SNL Figures 6.1-2 through 6.1-5 and 6.6-1 through 6.6-4 (2007bf). DOE numerical analyses indicate that substantial lateral diversion (associated with the PTn unit) away from the repository footprint may occur, which would reduce the amount of water passing through the footprint at the repository horizon (SAR Section 2.3.2.2.1.1).

The NRC staff notes that DOE adequately considered uncertainty in ambient mountain-scale flow patterns above the repository horizon in performance assessment calculations for the following reasons:

- The site-scale unsaturated zone flow model represents mountain-scale percolation fluxes as consistent with the DOE conceptual model and NRC staff understanding of the physical system. Further, DOE showed that alternative flow patterns with the same areal-average percolation within the repository footprint do not significantly affect repository performance.

- DOE showed that alternative representations for flow above the repository horizon tend to increase lateral diversion out of the repository footprint. Logically, diverting percolation from the repository footprint would reduce percolation fluxes through the repository horizon and thereby reduce calculated seepage estimates.

Site-Scale Unsaturated Zone Model Parameters

The NRC staff's risk-informed, performance-based review of the site-scale unsaturated zone model parameters includes consideration of (i) the technical bases for the parameters and (ii) the use of the parameters in the site-scale unsaturated zone model and downstream models used for performance assessment calculations. DOE does not base input parameters for the downstream seepage models on site-scale unsaturated zone model parameters; the NRC staff evaluates the seepage model input in TER Section 2.2.1.3.6.3.4.

DOE's site-scale unsaturated zone flow model represents the stratigraphy at Yucca Mountain using 32 homogeneous material property layers, with the same set of material properties assigned to every grid cell in a property layer (SAR Section 2.3.2.4.1.2.2). The material property layers were developed from geologic layers described in the Geologic Framework Model. DOE justified the use of layerwise homogeneous material properties on numerical simulations by comparing outputs from simulations with different levels of heterogeneity, concluding that similarities in fracture flux patterns and tracer transport times show that heterogeneities within units have only a minor effect on site-scale flow processes (SAR Section 2.3.2.4.1.1.4). DOE developed some model parameters directly from site observations (e.g., porosity) and some from model calibration to field measurements of saturation, potential, pneumatic pressure, and perched water elevations. DOE used a set of *in-situ* observations not used for model calibration (e.g., calcite, Carbon-14, and strontium) to build confidence in the flow model (SAR Section 2.3.2.5.1). DOE further used a set of *in-situ* observations not used for either model calibration or for building confidence in the flow model (i.e., chloride and temperature observations) to calibrate the uncertainty in infiltration (SAR Section 2.3.2.4.1.2.4.5).

The NRC staff notes from review of the SAR and supporting documentation, summarized in the preceding paragraph, that (i) the model parameters are based on site-specific information and (ii) DOE clearly documented the procedures and bases for developing the parameters.

The DOE performance assessment explicitly considered uncertainty in model parameters arising from uncertainty in infiltration flux, using a separate set of calibrated properties for the 10^{th}, 30^{th}, 50^{th}, and 90^{th} percentiles of the infiltration map uncertainty distribution. DOE ensured that the flow fields resulting from each set of calibrated properties featured predominantly vertical flow from the ground surface through the repository horizon within the repository footprint, in accordance with the DOE conceptual model. DOE did not propagate the full range of parameter uncertainties into the TSPA on the basis that sensitivity analyses demonstrating that the model results, including performance-affecting results such as flow pathways, are insensitive to the parameter values in the range that DOE considered (SAR Section 2.3.2.4.2.2).

The NRC staff notes that DOE adequately considers site-scale unsaturated zone model parameter uncertainty with respect to the Upper Natural Barrier representation in performance assessment calculations for the following reasons:

- The downstream feed for the site-scale unsaturated zone flow model as used in performance assessment calculations (percolation flux at the base of the PTn unit) is approximately equal to net infiltration under predominantly vertical flow.

- The Multiscale Thermal-Hydrologic Model (TER Section 2.2.1.3.6.3.3) and the near-field chemistry model (TER Section 2.2.1.3.3) use hydraulic parameters based on the site-scale unsaturated zone model.

Infiltration Uncertainty

The NRC staff evaluates infiltration uncertainty by considering (i) the technical bases for the DOE approach and (ii) consequences of the approach in performance assessment calculations. DOE addressed uncertainty in infiltration estimates both derived from the net infiltration model using input parameter uncertainty and from infiltration uncertainty from deep subsurface observations. The NRC staff evaluates the latter type of infiltration uncertainty in this section. DOE also referred to the results from incorporating this uncertainty as the unsaturated zone top (upper) boundary net infiltration to distinguish it from results derived from the net infiltration model.

For each climate state in the first 10,000 years after repository closure, DOE selected four infiltration maps calculated by the MASSIF infiltration model, out of the 40 equally likely realizations for the climate state (SNL, 2007az), to represent the uncertainty in infiltration. Initial probabilities of 0.2, 0.2, 0.3, and 0.3 were assigned to the four selected maps under each climate state, on the basis of the percentile rankings for the infiltration maps. Because the site-scale unsaturated zone model uses larger grid cells than the MASSIF model, DOE created upper boundary net infiltration maps for the site-scale unsaturated zone model grid. DOE transferred the infiltration values for each site-scale unsaturated zone model grid cell by accumulating infiltration fluxes from nearby MASSIF grid cells. For the post-10,000-year climate state, DOE assigned weights to the upper boundary net infiltration maps to be consistent with the draft rule describing deep percolation in the post-10,000-year period. The NRC staff evaluates DOE’s treatment of net infiltration during the post-10,000-year period in a subsequent subsection (Post-10,000-Year Approach).

DOE found that model predictions for temperature and chloride distributions in the unsaturated zone, using models based on the site-scale unsaturated flow model, did not reflect average measured quantities in the unsaturated zone. DOE used a version of the Generalized Likelihood Uncertainty Estimation methodology (Beven and Binley, 1992aa) to update the initial probability weights assigned to each flow field (SAR Section 2.3.2.4.1.2.4.5). The Generalized Likelihood Uncertainty Estimation methodology uses likelihood functions to revise initial probability weights on the basis of differences between field observations and numerical model predictions. A likelihood function is a statistical model that is used to describe how likely a population represents observed values. DOE updated the probability weights using temperature observations from 5 boreholes and chloride observations from 12 boreholes, the Exploratory Studies Facility, and the Enhanced Characterization for the Repository Block Cross Drift. DOE used four different likelihood functions to compare the model predictions with observations.

SAR Section 2.3.2.4.1.2.4.5.5 reported estimated final weighting factors of 0.619, 0.157, 0.165, and 0.0596 for the 10th, 30th, 50th, and 90th percentile upper boundary net infiltration maps, respectively, by combining the analyses considering temperature and chloride data. These final weights, when applied to the upper boundary condition infiltration maps of the ambient site-scale unsaturated zone model, result in weighted-average infiltration over the repository footprint of 8.46, 16.00, and 21.37 mm/yr [0.33, 0,63, and 0.84 in/yr] under present-day, monsoon, and glacial transition climate states as recorded in DOE Enclosure 1, Table 1 (2010ai). For comparison, the initial weights applied to the upper boundary condition infiltration maps result in weighted-average infiltration over the repository footprint of 17.31, 38.12, and 38.88 mm/yr [0.68, 1.50, and 1.53 in/yr] for the same climate states, also recorded in DOE Enclosure 1, Table 1 (2010ai). The values using the initial weights are larger than the

Generalized Likelihood Uncertainty Estimation-weighted infiltration by factors of 2.0, 2.4, and 1.8, respectively.

The NRC staff notes from review of the SAR and supporting documentation, summarized in the preceding paragraphs, that (i) the representation of infiltration uncertainty is based on site-specific information and (ii) DOE documented the procedures and bases for developing the infiltration uncertainty representation.

DOE identified several uncertainties in interpreting field observations using the temperature and chloride models (SAR Section 2.3.2.4.1.2.4.5.2). These uncertainties may affect the calculated probability weights for the assigned flow fields. DOE also showed that there is uncertainty associated with the likelihood functions used to determine the weights by showing that the calculated weights varied depending on the selection of a likelihood function (SAR Tables 2.3.2-25 through 2.3.2-27).

DOE considered expected doses in the first 10,000 years of performance for the seismic ground motion and igneous intrusion scenarios, which collectively account for approximately 97 percent of the calculated peak expected mean annual dose, as described in DOE Enclosure 5 (2009bo). DOE compared the performance assessment dose calculations of expected doses with and without the Generalized Likelihood Uncertainty Estimation weighting, finding that the original weighting scheme (without the Generalized Likelihood Uncertainty Estimation procedure) results in a 29 percent greater peak expected dose than the Generalized Likelihood Uncertainty Estimation weights. DOE concluded that, because the calculated doses are so similar, adjusting the infiltration uncertainty with the Generalized Likelihood Uncertainty Estimation procedure does not significantly affect performance assessment results.

The NRC staff notes that DOE's performance assessment calculations adequately represent the uncertainty in net infiltration for the following reasons:

- The NRC staff notes in TER Section 2.2.1.3.5 that DOE's representation of net infiltration obtained from the MASSIF net infiltration model is reasonable for use in performance assessment calculations.

- The NRC staff notes in a preceding subsection (Integration of Flow Models) that the MASSIF net infiltration model results are adequately integrated with the site-scale unsaturated zone flow model.

- DOE showed that the consequence of adjusting net infiltration uncertainties using the Generalized Likelihood Uncertainty Estimation procedure is inconsequential to repository performance.

Temporal and Spatial Variability in the Upper Unsaturated Zone

DOE described a primary role for the upper unsaturated zone (i.e., above the proposed repository) as strongly dampening and smoothing episodic infiltration pulses, to the extent that flows below the PTn within the proposed repository footprint are essentially steady, in equilibrium with the long-term climate. This is a screening argument for FEP 2.2.07.05.0A, Flow in the Unsaturated Zone (UZ) from Episodic Infiltration (SNL, 2008ab). DOE identified the consequence of this dampening effect as reducing time-averaged seepage rates into intact or degraded drifts, with the reduction effect becoming less significant as the drift degrades in DOE Enclosure 2 (2009an).

The NRC staff evaluates DOE's treatment of temporal and spatial variability by considering the potential consequences arising from partial breaches in the barrier capability represented by the upper unsaturated zone. The NRC staff identified effects on cooling, transport, and seepage as potential consequences resulting from a partial breach. The NRC staff focused on the consequences with respect to seepage, because in the DOE performance assessment calculations (i) conduction is the dominant thermal transport mechanism within the host rock, (ii) water moves from the repository horizon to the water table quickly relative to the performance period, and (iii) changes in percolation flux may have a nonlinear effect on seepage and release calculations. The NRC staff recognizes that temporal and spatial variability in percolation flux may nonlinearly affect release calculations in counteracting ways, because seepage may disproportionately increase with percolation. However, because of protection offered by the drip shield and waste package barriers, release rates may increase less than proportionately with seepage.

The NRC staff identified potential concerns with the DOE representation of temporal and spatial variability in percolation flux at the repository horizon for use in performance assessment stemming from (i) long-term (decadal to centennial) fluctuations around the mean climate, (ii) drift-scale percolation flux variability, and (iii) short-term (episodic or seasonal) fluctuations in deep percolation fluxes. DOE may not have considered the full range of uncertainty for these processes in its performance assessment calculations.

NRC staff considered the effects of DOE's representation of long-term fluctuations about the mean climate by considering the consequences on seepage. DOE considers decadal to centennial variability in percolation likely to occur below the PTn because the PTn has a finite storage capacity, but expects that the increase in calculated average seepage would be small if decadal to centennial variability was explicitly included. DOE showed that fluctuations in percolation flux of 20 and 50 percent about the mean (i.e., coefficients of variation of 0.2 and 0.5) yielded a systematic increase in Generalized Likelihood Uncertainty Estimation-weighted seepage of approximately 2.7 and 17 percent, respectively, under glacial transition conditions, as outlined in DOE Enclosure 1 (2009cc). Using an analysis demonstrating that a systematic increase in Generalized Likelihood Uncertainty Estimation-weighted seepage by a factor of approximately 2.5 has a negligible effect on the expected dose, as described in DOE Enclosure 5 (2009bo), DOE concluded by analogy that the smaller systematic increases in seepage induced by decadal to centennial climatic fluctuations also would have a negligible effect on performance assessment results. DOE reasonably represented long-term (decadal to centennial) climatic fluctuations in performance assessment calculations because DOE showed that such fluctuations have a negligible effect on the expected dose.

The NRC staff gains added confidence by using the approach in DOE Enclosure 5 (2009bo), along with independent analysis described in Stothoff Section 3 (2010aa) that considered a more extreme range of decadal- to centennial-scale fluctuations. Using a coefficient of variation of 0.8, which roughly corresponds to infiltration variability over a glacial cycle, as shown in Stothoff and Walter Table 4-2 (2007aa), the increase in seepage is smaller than the systematic increase in seepage that is attributed to the Generalized Likelihood Uncertainty Estimation approach, as detailed in DOE Enclosure 1 (2009cc).

DOE relied on numerical simulations to justify the representation of spatial variability in performance assessment calculations and to show that episodic flow is expected to be rare below the PTn. In these simulations, the NRC staff notes that DOE's calculated spatial variability in percolation is dependent on the estimated heterogeneity of input parameters, which DOE did not explicitly tie to all sources of spatial variability potentially indicated by site observations. DOE described several mechanisms to explain why some field evidence, such as modern tritium observations below the PTn, may not completely support the DOE assumptions related to drift-scale variability and episodic flow. These mechanisms include transport through faults, episodic flow, spatially variable infiltration patterns, and heterogeneous hydrologic properties, as outlined in DOE Enclosure 1 (2009cc).

To evaluate uncertainty in spatial variability, the NRC staff considered possible sources of spatial variability in percolation between mountain and drift scale that may not have been incorporated into the DOE models, such as intermediate-scale structural and stratigraphic features. The NRC staff considered the performance consequence of alternative representations of spatial variability using DOE's analyses reflecting intermediate-scale percolation flux variability. In those DOE analyses, the effect on seepage was considered for several alternative statistical relationships for the effect of rock heterogeneity on flow focusing, as shown in SNL Section 6.8.2, Case 6 (2007bf), ranging from no intermediate-scale focusing (Case 6a) to flow focusing that is more extreme than used for performance assessment calculations (Case 6c). DOE characterized the Case 6c distribution, which has a maximum flow-focusing factor more than six times larger than the distribution used in performance assessment calculations, as unrealistically extreme to represent rock heterogeneity, as described in DOE Enclosure 4 (2009bo). NRC staff treated Case 6c distribution as a surrogate that bounds spatial distribution not included in the DOE model. DOE showed that incorporation of the Case 6c distribution yields a minor increase in expected dose for the million-year seismic ground motion modeling case, as shown in DOE Enclosure 1 (2009cx). DOE described the seismic ground motion modeling case as the dominant contributor to calculated expected dose. Because DOE showed that the calculated peak cumulative areal-average releases are not strongly affected by an extreme increase in spatial variability, DOE adequately represented spatial variability between mountain and drift scales for performance assessment calculations.

The NRC staff evaluated the consequences of DOE's representation of flow as steady at the repository horizon, instead of episodic, by considering the field and modeling information DOE presented as a whole. DOE represented the PTn unit as a barrier with a strong potential to dampen episodic pulses below the PTn. This conceptual model was supported by numerical modeling results interpreting chloride, temperature, and radioisotope data obtained from within and below the PTn. DOE estimated that approximately 1 percent of the repository is affected by fast pathways, predominantly in faults (SAR Section 2.3.2.2.1.1), and argued that these fast pathways are not necessarily a consequence of episodic flow, but instead would represent spatial variability. The NRC staff notes that the information DOE presented (i) shows that the PTn has the potential to strongly dampen episodic pulses below the PTn and (ii) suggests that any areas at the repository horizon that exhibit episodic pulses are likely to require a combination of flow focusing and fast pathways through the PTn to overcome the PTn dampening potential. As a consequence, DOE's data and modeling results suggest that, if episodic pulses do penetrate the PTn, relatively few waste packages are likely to be affected by these episodic pulses. Therefore, the NRC staff notes that the effect of episodic percolation can be evaluated by an analysis of the effect of the spatial extent of fast pathways that breach the barrier capability of capillary diversion at the drift (i.e., seepage becomes equal to percolation). The NRC staff considered potential consequences of episodic flow using information in DOE Enclosure 1, Tables 4 and 5 (2010ai) to perform a calculation that assumes an extreme representation of the extent of episodic flow, as described in Stothoff Section 3 (2010aa). In the NRC staff analysis, staff assumes that (i) episodic flow completely avoids the capillary barrier and becomes seepage, (ii) episodic flow pathways occur under the largest fluxes, and (iii) the area experiencing fast pathways is five times larger than DOE estimated. Each of these assumptions conservatively represents an aspect of the DOE seepage abstraction. Under these assumptions, all of DOE's Percolation Bin 5 (highest flux bin) is modeled as completely ineffectual at preventing seepage (seepage rates are equal to percolation rates). For comparison, seepage rates are less than 7 percent of infiltration rates for Percolation Bin 5 in the DOE model under glacial-transition conditions. With these assumptions, NRC staff calculates an areal-average seepage rate 2.4 to 3 times larger than DOE used in performance assessment calculations, which is less than the 3.9 times increase in total seepage that DOE calculated in its analysis of the Case 6c flow-focusing distribution. As previously stated, DOE showed that this order of increase in total seepage does not significantly increase calculated dose.

The NRC staff notes that repository performance is not strongly affected by the DOE representation of episodic flow for performance assessment calculations because (i) DOE showed that repository performance is not strongly affected by increased seepage from increased infiltration or increased spatial variability and (ii) NRC independent analysis of an extreme case showed that the increase in seepage that might occur from episodic flow results in a similarly small increase in seepage.

In summary, DOE reasonably represented temporal and spatial variability within the upper unsaturated zone because repository performance with respect to expected dose is not strongly affected by alternative representations.

Post-10,000-Year Approach

DOE chose not to explicitly model climate or infiltration for the post-10,000-year period, and instead used a distribution of percolation rates. To derive its distribution, DOE (i) selected four of the upper boundary condition net infiltration maps used to represent specific climate states in the first 10,000 years of performance; (ii) modified these maps into four new maps to achieve areal-average deep percolation flux target values within the repository footprint, at the infiltration-map probability values used in the first 10,000 years after closure; and (iii) created four site-scale unsaturated zone model flow fields on the basis of these boundary conditions (SAR Section 2.3.2.4.1.2.4.2).

DOE used the percolation distribution in proposed 10 CFR Part 63 (SAR Section 2.3.2.4.1.2.4.2) because the final rule was not promulgated until a few months before DOE submitted the SAR. Reflecting the difference between the draft and final percolation distributions, the mean percolation in the final rule is 16 percent larger than that in the proposed rule. In DOE Enclosure 6 (2009cb), DOE performed sensitivity analyses that showed no significant affect on repository performance when using the distribution from the proposed rule instead of that from the final rule.

The NRC staff notes that DOE reasonably considered the prescribed deep percolation rates and distributions for the following reasons:

- DOE evaluated repository performance using a probability distribution of areal-average deep percolation fluxes that adequately reflects the prescribed distribution.

- The upper boundary condition net infiltration maps and deep percolation fluxes for the post-10,000-year period are based on FEPs considered in the first 10,000 years of performance consistent with NRC (2009ab).

- DOE represents infiltration uncertainty in the first 10,000 years of performance and in the post-10,000-year period using a consistent approach.

The NRC staff evaluated the spatial distribution of net infiltration results for the post-10,000-year period by considering the potential effects of features and processes affected by climate change, such as alterations in soil depth, soil profile development, and changes in caliche volume. The NRC staff discusses in TER Section 2.2.1.3.5 the influence of both temporal and spatial aspects of features and processes on the spatial distribution of net infiltration results. Therefore, the effect of changes to properties over time can be evaluated in terms of alternative spatial distribution of net infiltration. The NRC staff noted, as described in the previous section of this chapter, that DOE's performance assessment calculations would not be strongly affected by including systematic changes in seepage resulting from alternative representations of spatial variability, long-term climatic variability, and episodic fluctuations during the first 10,000 years of performance. The NRC staff reasons that these effects would be no more important in the post-10,000-year period because (i) these effects have a lesser effect on DOE's seepage model calculations as percolation fluxes increase, and the specified percolation fluxes in the post-10,000-year period are generally larger than those used during the first 10,000 years and (ii) these effects have a lesser effect on seepage for degraded drifts, and DOE includes drift degradation from accumulated seismic events in performance assessment calculations for the post-10,000-year period.

Summary

The NRC staff reviewed the model conceptualization, the underlying assumptions of the ambient site-scale unsaturated zone flow model and other relevant abstractions with which the ambient site-scale unsaturated zone flow model exchanges data and information, and the alternative model conceptualizations DOE used to analyze model uncertainties. The NRC staff notes that

- DOE considered the important flow-affecting processes and provided adequate technical bases for their inclusion in the abstracted ambient site-scale unsaturated zone flow model and downstream abstractions used in performance assessment calculations

- DOE integrated intermediate-scale and fine-scale flow processes to represent the Upper Natural Barrier

- DOE represented ambient mountain-scale percolation fluxes above the repository horizon and ambient drift-scale percolation fluxes at the repository horizon

- DOE represented spatial and temporal variability of model parameters and boundary conditions to estimate flow paths and percolation flux with respect to the Upper Natural Barrier

- DOE provided sufficient data for ambient site-scale unsaturated zone model justification

- DOE characterized and propagated ambient site-scale unsaturated zone model parameter uncertainty in process-level and performance assessment models with respect to the Upper Natural Barrier

- DOE considered percolation during the period from 10,000 to 1 million years after permanent closure

2.2.1.3.6.3.3 Thermohydrologic Effects of Waste Emplacement

DOE represented the unsaturated zone and Engineered Barrier System using a hierarchy of far-field, near-field, and within-drift models to account for thermal effects due to emplacement. DOE used a conceptual and numerical model, the Multiscale Thermal-Hydrologic Model, to represent near-field and in-drift thermohydrologic conditions. DOE used input from the site-scale unsaturated zone flow model to account for far-field effects and output from the Multiscale Thermal-Hydrologic Model in downstream models that estimate the effects of in-drift thermohydrologic conditions on thermal seepage, quantity and chemistry of water contacting engineered barriers and waste forms, degradation of engineered barriers, and radionuclide release rates and solubility limits. This TER section describes the NRC staff evaluation of the effects of waste emplacement on near-field and in-drift thermohydrologic conditions. The NRC staff evaluated repository performance related to the effects of thermal load on (i) seepage, (ii) quantity and chemistry of water contacting engineered barriers and waste forms, (iii) degradation of engineered barriers and radionuclide release rates, and (iv) solubility limits in TER Sections 2.2.1.3.6.3.4, 2.2.1.3.3, 2.2.1.3.1, and 2.2.1.3.4, respectively.

DOE passed Multiscale Thermal-Hydrologic Model output to several downstream models in its performance assessment calculations. DOE used drift-wall temperature to switch between two limiting conditions for seepage, assuming that (i) no seepage occurs where drift-wall temperature is greater than 100 °C [212 °F] {the boiling temperature of water at the repository elevation is approximately 96 °C [205 °F]} and (ii) seepage occurs at the ambient rate for lower drift-wall temperatures (SAR Section 2.3.3.4). DOE modeled waste-package corrosion rates as depending on waste-package temperature and relative humidity, and temperature-dependent chemistry of seeping water (SAR Section 2.3.6). DOE modeled a diffusive-release pathway forming within failed waste packages once a continuous liquid film forms at elevated waste-package relative humidity levels. DOE used invert saturation, invert temperature, and imbibition fluxes into the invert to estimate properties and fluxes affecting released radionuclide transport from the waste package to the host rock (SAR Section 2.3.7).

The DOE repository design basis places limits on the (i) peak waste package temperature {300 °C [572 °F] for 500 years, followed by 200 °C [392 °F] for 9,500 years} to reduce the potential for degradation of Alloy 22 waste packages; (ii) peak postclosure drift wall temperature {200 °C [392 °F]} to reduce thermal effects on drift stability; and (iii) peak mid-pillar temperature {96 °C [205 °F]} to facilitate drainage of percolation water between emplacement drifts (SAR Section 2.3.5.4.3; SAR Table 1.3.1-2). DOE's analysis indicated that the peak temperature limits can be accomplished through thermal loading criteria for waste packages (SAR Section 1.3.1.2.5). DOE used the Multiscale Thermal-Hydrologic Model to show that these design basis temperature limits can be achieved using (i) a stylized postclosure reference case based on expected waste package receipts over the emplacement period and (ii) two estimated limiting waste streams developed with different management options (SAR Section 1.3.1.2.5).

In evaluating repository performance with respect to near-field and in-drift thermohydrologic conditions, the NRC staff reviews (i) the DOE conceptual model, (ii) the process-level implementation of the DOE conceptual model, (iii) data support and propagation of uncertainty, (iv) abstraction of the process-level model into performance assessment calculations, and (v) use of the model outputs in downstream models. The NRC staff's review of these topics focuses on aspects of repository performance that affect (i) duration of above-boiling temperatures at the drift wall, (ii) waste-package temperatures, (iii) in-drift humidity after onset of seepage, and (iv) seepage fluxes into inverts. These aspects of the Multiscale Thermal-Hydrologic Model are outputs that are used as input for downstream models that calculate seepage, corrosion of waste packages, and release pathways within the Engineered Barrier System. The NRC staff review also focuses on the DOE representation of collapsed drifts, because burial of waste packages by an insulating rubble layer could result in elevated waste-package temperatures.

Conceptual and Implemented Numerical Models

NRC staff compared the DOE description of the conceptual model for thermohydrologic effects of waste emplacement provided in SAR Sections 2.3.3.1.1 and 2.3.5.4 (and selected references) with the NRC staff understanding of the Yucca Mountain natural system, obtained from extensive prelicensing experience and independent analyses of thermohydrologic processes at Yucca Mountain (e.g., NRC, 2005aa; Painter, et al., 2001aa; Manepally, et al., 2004aa). The NRC staff notes that DOE adequately described its conceptual model of the thermohydrologic effects of waste emplacement, because DOE described (i) the features of the site geology and engineered barriers that affect thermohydrologic processes and (ii) the most important thermohydrologic processes and parameters affecting the performance of the upper natural barrier, such as in-drift temperature and relative humidity, host rock drying and rewetting, thermal conduction, and thermal radiation.

DOE implemented the conceptual model into the four submodels of the Multiscale Thermal-Hydrologic Model. The submodels were linked through the mathematical principle of superposition, representing different aspects of coupled thermohydrologic processes at different spatial scales (SAR Section 2.3.5.4.1.3.1). These submodels consider (i) three-dimensional mountain-scale conduction, (ii) two-dimensional drift-scale thermal-hydrology in cross sections (“chimneys”) perpendicular to the drift axis, (iii) links between the mountain-scale and chimney submodels, and (iv) effects of discrete waste packages. On the basis of the descriptions supplied in SAR Section 2.3.5.4 and SNL (2008aj), the NRC staff notes that, with respect to each submodel, DOE (i) reasonably documented the procedures used to develop and support the submodel, (ii) developed the submodels of the Multiscale Thermal-Hydrologic Model in accordance with the corresponding conceptual model for the submodel, and (iii) used standard numerical approaches for the thermohydrologic processes considered in the submodel. The NRC staff notes that while DOE did not transparently describe the overall Multiscale Thermal-Hydrologic Model methodology linking the submodels, it reasonably showed that the linked submodels of the Multiscale Thermal-Hydrologic Model adequately represent thermohydrologic processes using comparisons to alternative models. The NRC staff gained confidence in the completeness and representativeness of the DOE description of the technical basis for the Multiscale Thermal-Hydrologic Model, as shown in SAR Section 2.3.5.4 and SNL Section 6.2 (2008aj), by comparing the technical basis with the NRC staff understanding of numerical simulation approaches, obtained from extensive prelicensing experience and independent analyses of thermohydrologic processes at Yucca Mountain (e.g., NRC, 2005aa; Painter, et al., 2001aa; Manepally, et al., 2004aa).

DOE used its Drift Scale Test to validate the conceptual model underlying the drift-scale thermal-hydrologic submodel, as described in SAR Section 2.3.5.4.1.3.3 and SNL Section 7.4 (2008aj), and used its Large Block Test, shown in SNL Section 7.3 (2008aj), to build confidence in the ability of the Multiscale Thermal-Hydrologic Model to predict thermal-hydrologic processes in the host rock. DOE (i) simulated the thermohydrologic behavior observed in the

Drift Scale Test and Large Block Test using the same modeling techniques included in the thermohydrologic submodel of the Multiscale Thermal-Hydrologic Model; (ii) compared modeled and observed temperature, relative humidity, and liquid-phase saturation values; and (iii) concluded that the differences were within the parametric uncertainty of Multiscale Thermal-Hydrologic Model results. On the basis of these considerations, the NRC staff notes that DOE showed that thermohydrologic processes observed in these heater tests are adequately included in Multiscale Thermal-Hydrologic Model.

DOE identified several assumptions and limitations associated with linking the submodels (SAR Section 2.3.5.4.1.3.1), including restrictions related to mountain-scale and along-drift convection. DOE considered alternative conceptual models by comparing Multiscale Thermal-Hydrologic Model results with those of (i) an east-west cross section from a smeared-heat-source mountain-scale model; (ii) a three-dimensional, mountain-scale, nested-grid thermal-hydrologic model for a three-drift test case; and (iii) a three-dimensional, pillar-scale model with different axial in-drift vapor transport assumptions (SAR Section 2.3.5.4.1.3.3). The NRC staff notes that (i) DOE provided reasonable information to assess the model implementation uncertainty, because these comparisons range from mountain-scale to detailed pillar-scale simulations, and (ii) DOE used a modeling approach that conservatively represents the effects of in-drift vapor transport on in-drift relative humidity, because in-drift axial transport would tend to systematically delay rewetting compared to the Multiscale Thermal-Hydrologic Model results. Therefore, DOE reasonably showed that the Multiscale Thermal-Hydrologic Model results are comparable to alternative conceptual models in the intended thermal regime for repository operation. DOE assumed that mobilized water predominantly moves perpendicular to the axis of the emplacement drifts within the host rock. NRC staff notes this is consistent with the intended use of the Multiscale Thermal-Hydrologic Model under the design thermal regime for repository operation, because mobilized water can drain within the mid-pillar region without systematic mountain-scale lateral redistribution of water when boiling conditions are not reached throughout the pillar.

In summary, DOE adequately provided the technical bases for the individual submodels and the linkages among the submodels supporting the intended use of the Multiscale Thermal-Hydrologic Model under the design thermal regime for repository operation, as discussed previously.

Abstraction of the Multiscale Thermal-Hydrologic Model for TSPA

The Multiscale Thermal-Hydrologic Model calculates time-dependent thermal-hydrologic variables for each of the 8 waste packages simulated for the 3,264 subdomains, each representing a 20-m [66-ft] segment of an emplacement drift. DOE referred to this as the comprehensive set of Multiscale Thermal-Hydrologic Model outputs. DOE performed the calculations for 7 parameter uncertainty cases, representing 12 combinations of infiltration uncertainty and thermal conductivity uncertainty, as described in the next subsection. DOE mapped each of the 3,264 subdomains to one of the five percolation bins abstracting the effects of seepage.

DOE provided the comprehensive set of Multiscale Thermal-Hydrologic Model outputs for waste package temperature and relative humidity for different waste package types and drift-wall temperature, for all Multiscale Thermal-Hydrologic Model subdomains in each percolation bin, as input to the Waste Package Degradation Model Component, the Drift Seepage Submodel, and the Drift Wall Condensation Submodel within the TSPA model, as shown in SNL Section 6.3.2.2 (2008ag). These downstream models use Multiscale

Thermal-Hydrologic Model outputs to calculate waste package failure rates. The NRC staff notes that DOE abstracts Multiscale Thermal-Hydrologic Model output for these downstream models because the entire set of process-level output is provided to these downstream models.

DOE also abstracted the Multiscale Thermal-Hydrologic Model results by selecting a single representative codisposal waste package and commercial spent nuclear fuel waste package for each percolation bin. DOE used the codisposal and commercial spent nuclear fuel waste packages with peak waste package temperature and drift-wall boiling duration closest to the median values to represent thermohydrologic conditions for all waste packages in the percolation bin. DOE provided the time-dependent output values of waste package surface temperature and relative humidity, drift wall temperature, invert temperature, invert saturation, and flux into the invert for the selected representative waste packages to the Waste Form Degradation and Mobilization Model Component, the Engineered Barrier System Flow and Transport Model Component, the Engineered Barrier System Chemical Environment Submodel, and the Drift Wall Condensation Submodel within the TSPA model, as outlined in SNL Section 6.3.2.2 (2008ag). These downstream models use the Multiscale Thermal-Hydrologic Model outputs to calculate radionuclide release rates from failed waste packages and radionuclide transport within the Engineered Barrier System. DOE provided analyses in SNL Section 7.3.4.3.1 (2008ag) comparing estimates of cumulative radionuclide releases from a single failed waste package using the representative location with estimates calculated using the comprehensive set of Multiscale Thermal-Hydrologic Model output. These analyses showed essentially identical cumulative release of representative radionuclides from the Engineered Barrier System after both 10,000 and 1 million years. DOE reasonably abstracted Multiscale Thermal-Hydrologic Model process-level results in performance assessment calculations with respect to these downstream models because DOE showed that the abstraction used for performance assessment calculations does not lead to an underestimate of radionuclide releases from the Engineered Barrier System.

To provide further confidence in the abstraction process, the NRC staff considered whether uncertainty in heterogeneity was adequately incorporated into the comprehensive set of Multiscale Thermo-Hydrologic Model results. The NRC staff used systematic changes to the duration of boiling as a key thermal-hydrologic measure of performance consequence, because a shorter average boiling duration would result in earlier seepage and an earlier onset of enhanced release from failed waste packages. DOE identified host rock thermal conductivity and percolation flux as the dominant parameters responsible for variability and uncertainty in simulated thermal hydrologic conditions (SAR Section 2.3.5.4.1.3.2), but did not explicitly represent local heterogeneity in host rock thermal conductivity or drift-scale percolation flux variability in Multiscale Thermal-Hydrologic Model calculations. The NRC staff expects that local variability in temperature would be increased if such heterogeneity were included. Because local variability logically leads to approximately the same number of waste packages experiencing temperature increases and decreases, and therefore does not systematically shorten the average boiling duration, the NRC staff gained confidence that DOE represented uncertainty and variability in input parameters for the abstraction of Multiscale Thermal-Hydrologic Model results.

Data Support and Uncertainty Propagation in the Multiscale Thermal-Hydrologic Model

NRC staff reviewed the information provided in SAR Section 2.3.5.4.1.2 and selected references to evaluate the DOE supporting data and characterization of uncertainties in the Multiscale Thermal-Hydrologic Model. Because of the importance of the following three topics in performance assessment calculations, the NRC staff focuses its review on (i) input parameters that significantly affect Multiscale Thermal-Hydrologic Model results, (ii) data support for uncertainty bounds in input parameters, and (iii) uncertainty propagation in the Multiscale Thermal-Hydrologic Model.

Multiscale Thermal-Hydrologic Model simulations require input to describe the Engineered Barrier System and natural barriers (SAR Section 2.3.5.4.1.2.2). DOE identified the design control parameters and associated design constraints in SAR Table 2.2-3. DOE derived Engineered Barrier System parameters from the design information. DOE based natural barrier parameters on the site-scale unsaturated zone flow model (evaluated in TER Section 2.2.1.3.6.3.3), including hydrologic properties of the unsaturated zone, natural system geometry, and percolation fluxes below the PTn.

DOE explicitly represented the drift, drip shield, and invert components of the Engineered Barrier System {SNL Section 4[a] (2008aj)}. DOE screened out representing the thermohydrologic behavior of other engineered components potentially affecting performance (e.g., rock bolts and associated boreholes used for ground support) on the basis of low consequence for performance assessment calculations. The NRC staff evaluates DOE's screening arguments with respect to FEP 1.1.01.01.0B (Influx through Holes Drilled in Drift Wall or Crown) and 2.1.06.04.0A (Flow Through Rock Reinforcement Materials in Engineered Barrier System) in TER Section 2.2.1.2.1.1, finding both screening arguments reasonable, based on the small effect on performance assessment calculations. DOE described the design information representing engineered features in the Multiscale Thermal-Hydrologic Model as consistent with the design of subsurface structures, systems, and components (SAR Section 2.3.5.4.1.2.1). In SNL Section 4.1 (2008aj), DOE described repository subsurface and waste package design information as obtained from controlled sources and based on the current repository design.

NRC staff notes that DOE adequately represented the input parameters describing engineered components with respect to the thermohydrologic parameters the Multiscale Thermal-Hydrologic Model calculated because DOE (i) identified the Engineered Barrier System components predominantly influencing in-drift thermohydrologic conditions, (ii) included these components in the Multiscale Thermal-Hydrologic Model, (iii) reasonably screened out other components, and (iv) described the included Engineered Barrier System components using parameter values consistent with the current repository design.

DOE identified host rock thermal conductivity and percolation flux as the dominant parameters responsible for variability and uncertainty in simulated thermal-hydrologic conditions (SAR Section 2.3.5.4.1.3.2). NRC staff evaluated the adequacy of percolation flux in TER Section 2.2.1.3.6.3.2; incorporation of uncertainty in percolation is described in the paragraphs that follow. DOE considered the uncertainty in thermal conductivity of the host rock using a geostatistical model supported by laboratory measurements and core samples to constrain and condition the geostatistical model (BSC, 2004bf). DOE extracted host rock thermal conductivity values from the geostatistical model, evaluated the influence of thermal conductivity on peak waste package temperatures and duration above boiling, and assigned weights for implementation in TSPA, as outlined in SNL Section 6.2.13.3[a] (2008aj). DOE averaged the thermal properties of nonrepository units to facilitate computational efficiency, using a sensitivity analysis to show that the averaging does not affect the Multiscale Thermal-Hydrologic Model results, as shown in SNL Section 6.2.13.4[a] (2008aj). The percolation flux applied at the top boundary in the thermohydrologic submodel is the percolation flux at the base of the PTn unit calculated in the site-scale unsaturated zone flow model for the nominal 10^{th}, 30^{th}, 50^{th}, and 90^{th} percentile scenarios. The uncertainty in percolation flux in the site-scale unsaturated zone flow model at the mountain scale is propagated consistently in the Multiscale

Thermal-Hydrologic Model. The NRC staff notes that DOE reasonably propagated the uncertainty in host rock thermohydrologic properties with respect to observations into Multiscale Thermal-Hydrologic Model input, because DOE appropriately identified the dominant thermohydrologic property affecting uncertainty and used reasonable methods to propagate the uncertainty from the available laboratory and field data into the Multiscale Thermal-Hydrologic Model input.

DOE propagated uncertainty in host rock thermal conductivity and percolation flux into simulations using 7 of the 12 combinations of 10^{th}, 30^{th}, 50^{th}, and 90^{th} percentile flux scenarios with low, mean, and high thermal conductivities as input (SAR Section 2.3.5.4.1.3.2). For each of the remaining five combinations, DOE used the results from one of the seven simulations as a surrogate on the basis of similarity in boiling duration, as described in SAR Section 2.3.5.4.1.3.2 and SNL Section 6.2.12.3[a] (2008aj). DOE propagated the full set of 12 combinations into performance assessment calculations. The NRC staff notes that DOE reasonably propagated uncertainty in thermal-hydrologic conditions using the key performance-affecting result (duration of drift-wall boiling) to select surrogate simulations and considering the full range of the key thermal-hydrologic parameters.

In summary, the NRC staff notes that DOE reasonably propagated uncertainty in thermohydrologic parameters into performance assessment calculations because

- DOE identified the key performance-affecting thermohydrologic parameters and factors

- DOE adequately propagated uncertainty from observations into Multiscale Thermal-Hydrologic Model input

- DOE represented uncertainty in the key performance-affecting parameters using appropriate combinations of the uncertain parameters

Use of Multiscale Thermal-Hydrologic Model Results in Downstream Models

NRC staff evaluated predictions of thermohydrologic conditions in part by considering how DOE used Multiscale Thermal-Hydrologic Model results in downstream models. The downstream models using Multiscale Thermal-Hydrologic Model results include (i) thermal seepage, (ii) quantity and chemistry of water contacting engineered barriers and waste forms, (iii) degradation of engineered barriers, and (iv) radionuclide release rates and solubility limits. The NRC staff evaluates these downstream models in TER Sections 2.2.1.3.6.3.4, 2.2.1.3.3, 2.2.1.3.1, and 2.2.1.3.4, respectively.

On the basis of the NRC staff review, the NRC staff notes that DOE properly used the Multiscale Thermal-Hydrologic Model results and parameters in downstream models because they are generally used in a manner consistent with Multiscale Thermal-Hydrologic Model assumptions. DOE chose to not represent the full range of local variability in Multiscale Thermal-Hydrologic Model results in some downstream model calculations. On the basis of DOE sensitivity analyses [e.g., SNL Section 6.3 (2008aj)], the NRC staff notes that DOE reasonably represented local variability in Multiscale Thermal-Hydrologic Model results (e.g., variability in drift wall temperature, waste package temperature and relative humidity, imbibition fluxes) because DOE showed that local variability on radionuclide releases minimally affects duration or magnitude of radionuclide releases.

Uncertainty in Thermal Loading

DOE described the loading strategy implemented in the Multiscale Thermal-Hydrologic Model as a stylized postclosure reference case based on expected waste package receipts over the emplacement period (SAR Section 1.3.1.2.5). DOE considered the actual future waste stream to be uncertain and considered flexibility in emplacement strategies necessary to manage acceptance of a wide spectrum of waste streams (SAR Section 1.3.1.2.5). For example, the design heat load DOE described in SNL (2008ai) updated the heat load used for Multiscale Thermal-Hydrologic Model calculations (SAR Section 2.3.5.4.1). In SAR Section 2.3.5.4.3, DOE described how temperature estimates using the design heat load compared with the temperature estimates from the Multiscale Thermal-Hydrologic Model results. DOE stated that it would develop an emplacement drift plan [SAR Section 1.3.1.2.5; DOE Enclosure 1 (2009ct)] for each drift, or set of drifts, that will (i) provide specific information, such as waste characteristics, waste package emplacement locations, and ventilation duration, and (ii) describe how preclosure and postclosure performance criteria will be met using the selected emplacement strategy.

DOE showed that management approaches using temperature index functions representing three- and seven-package segments are capable of achieving performance targets for mid-pillar, drift-wall, and waste-package peak temperatures (SAR Section 2.3.5.4.3) for the two estimated limiting waste streams. The management approaches utilize extended duration of ventilation beyond 50 years or different surface handling facilities and aging capacities. DOE concluded that (i) only minor modifications to the TSPA model inputs are needed to represent the anticipated range of thermal loading; (ii) the geomechanical, hydrogeologic, and geochemical system responses for the two estimated limiting waste streams are each within the range of applicability for the respective models, as shown in SNL Section 6.4 (2008ai); and (iii) the changes in system responses arising from future waste streams different than the reference case do not significantly affect the screening justifications for excluded FEPs or the modeling basis for included FEPs, as outlined in SNL Section 6.5 (2008ai). The NRC staff notes that DOE showed that practical management strategies can be devised to achieve postclosure design basis targets in intact drifts using surface aging (limited to the 50-year emplacement period) and flexibility in ventilation duration beyond the minimum length of 50 years.

DOE expects that, under bounding assumptions, peak waste package temperatures for some waste packages may exceed the design basis temperature by nearly 100 °C [180 °F] if drifts were to collapse within the first 90 years after closure (SAR Section 2.3.5.4.3, SAR Figure 2.3.5-37). DOE considered several mechanisms for drift collapse, including seismic-induced ground motion, thermally induced stresses, and gravitational stresses. DOE screened out all mechanisms for drift collapse other than seismic-induced ground motion (FEP 2.1.07.02.0A, Drift Collapse) on the basis of low consequence (SNL, 2008ab). The NRC staff, in TER Section 2.2.1.2.1.3.2 (FEP 2.1.07.02.0A), notes that DOE provided an adequate basis for the screening decision. DOE screened in mechanisms of drift degradation from seismic-induced ground motion, including in-drift temperature and relative humidity consequences from seismic-induced drift collapse. DOE performed a bounding probabilistic risk analysis considering uncertainty in seismic events and the key thermohydrologic parameter (thermal conductivity). This risk analysis used methods and assumptions consistent with DOE's performance assessment calculations to calculate a probability of approximately 1 in 10,000 that the hottest waste package in the stylized postclosure reference case exceeded the 300 °C [572 °F] waste package temperature design basis because of drift collapse during the first 10,000 years after closure, as described in SAR Section 2.3.5.4.3, SNL Section 6.3.17[a]

(2008aj), and SNL Section 6.4.2.5 (2008ai). DOE screened out consideration of peak waste package temperatures exceeding the established temperature limits in performance assessment calculations on the basis of the probabilistic calculation and additional information describing the nature of the bounding assumptions (SAR Section 2.3.5.4.3). DOE adequately justified screening out the risk from Engineered Barrier System temperatures exceeding the established temperature limits as a result of drift collapse because DOE (i) used appropriate methods consistent with performance assessment calculations to develop risk estimates and (ii) applied an approximate screening criterion for the probability of occurrence.

The NRC staff recognizes that actual waste sent to Yucca Mountain, flexible emplacement strategies, and natural system and modeling factors may increase or decrease estimated peak temperatures relative to the stylized reference case [e.g., SNL Figure 6.4.2-28 (2008ai)], which may result in locally different thermal regimes compared to the performance assessment and screening calculations. DOE stated that it would develop an emplacement drift plan prior to waste emplacement that specified for each drift, or set of drifts, the (i) waste characteristics, (ii) waste package emplacement locations, (iii) ventilation duration, and (iv) how preclosure and postclosure performance criteria will be met using the selected emplacement strategy, as described in SAR Section 1.3.1.2.5 and DOE Enclosure 1 (2009ct). To confirm the heat load is in the range analyzed in the SAR, DOE should provide emplacement loading plans before emplacement begins.

Summary

In this TER section, the NRC staff evaluated the in-drift and near-field thermohydrologic conditions estimated using the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4.1). With respect to the Multiscale Thermal-Hydrologic Model, DOE adequately described the (i) conceptual models, (ii) implementation, (iii) performance assessment abstraction, (iv) data used to derive inputs, and (v) uses of output in performance assessment calculations. DOE adequately considered (i) conceptual model uncertainty, (ii) alternative conceptual models, and (iii) data variability and uncertainty with respect to the application of the Multiscale Thermal-Hydrologic Model for performance assessment calculations. On the basis of the previous discussion, the NRC staff notes that DOE adequately integrated the Multiscale Thermal-Hydrologic Model with downstream models in performance assessment calculations. The NRC staff notes that DOE adequately demonstrated the capability for devising emplacement loading strategies that achieve postclosure design basis targets. To confirm the heat load is in the range analyzed in the SAR, DOE should provide emplacement loading plans before emplacement begins.

2.2.1.3.6.3.4 Ambient and Thermal Seepage Models

This section contains the NRC staff review of DOE’s model and results for water seeping into drifts. Seepage into drifts encompasses a subset of processes in the unsaturated flow system that occurs in the vicinity of the drift wall. DOE described seepage as a component of the second feature, the unsaturated zone above the repository, within the Upper Natural Barrier (SAR Section 2.1.2.1). DOE considered seepage (SAR Section 2.3.3) separately from unsaturated zone flow (SAR Section 2.3.2) because of the smaller scale of analysis needed for the processes important for seepage and, consequently, the need for a different set of data and models to produce results for use in the performance assessment.

DOE strictly defined seepage as liquid water that drips from the drift ceiling and thus could potentially contact engineered barrier components. Two primary processes provide barrier capability in DOE's seepage model: capillary diversion of liquid water around large openings (drifts in this case) and vaporization in the host rock that creates a dry zone around the drifts (SAR Section 2.1.2.1). Capillary forces may make drifts barriers to flow by inducing water to laterally flow (divert) around the large opening. During the thermal period, the vaporization barrier refers to the boiling of water in the host rock and migration of the resultant vapor to locations away from the heat source. In the DOE abstraction, the resultant creation of a dryout zone surrounding a drift leads to elimination of liquid flux at the drift wall.

Three inputs are provided to the seepage abstraction from other areas of the natural systems. First, the Multiscale Thermal-Hydrologic Model passes the distribution of percolation rates across the repository to the seepage abstraction (SAR Section 2.3.5.4.1). The values used are consistent with those from the ambient, site-scale unsaturated zone model (SAR Section 2.3.2). Second, the Multiscale Thermal-Hydrologic Model passes the temperature history for the drift wall to the seepage abstraction (SAR Section 2.3.5.4.1). Third, the thermal-mechanical model abstraction (SAR Section 2.3.4) passes the accumulated amount of rubble to the seepage abstraction, which uses it to reflect the degradation state of drift openings for seepage calculations.

The DOE seepage abstraction passes two outputs to the Engineered Barrier System models: the seepage rate and the fraction of drift segments where liquid water seeps into drifts. Drift segments can be thought of as waste package locations. The total dripping flux (SAR Section 2.3.3), which is the sum of the seepage and condensation flux (SAR Section 2.3.5.4.2), is the flux of liquid water leaving the drift wall and contacting engineered components. NRC staff reviews DOE estimates of condensation flux in TER Section 2.2.1.3.6.3.5.

In its review of the DOE seepage estimates, NRC staff considered how and to what extent seepage affects performance. The fraction of waste package locations getting wet and seepage flux in those wet areas are both passed directly from the DOE seepage abstraction model to the Engineered Barrier System models. The condition of the drip shields (intact or degraded) plays a critical role in the DOE performance assessment. When intact, drip shields prevent liquid water from contacting the waste package. In this case, seepage has no influence on the release of radionuclides, and transport rates out of failed waste packages are constrained to the slower diffusive rate of radionuclide movement rather than the faster advective rate. When degraded, drip shields do not divert all water away from waste packages. In this case, the Engineered Barrier System models use seepage estimates, which may influence the (i) corrosion of engineered components; (ii) number of waste packages contacted by water; and (iii) dissolution, mobilization, and transport of radionuclides to the unsaturated zone below the drifts. NRC staff reviews these three areas, which include processes and features from the drip shield to the invert/host rock interface, in TER Sections 2.2.1.3.1, 2.2.1.3.3, and 2.2.1.3.4.

Consistent with a risk-informed approach and NRC guidance (NRC, 2003aa), the remainder of this seepage section focuses on review of the information and bases DOE provided for the (i) development of the ambient seepage abstraction; (ii) capillary diversion for intact drifts; (iii) capillary diversion for degraded drifts; (iv) seepage fraction, which is the fraction of repository that is in the seeping environment; (v) spatial variability of flow; and (vi) thermal seepage.

Development of Ambient Seepage

This section reviews DOE's description of seepage processes, field tests, and measured data and how they are incorporated into the seepage abstraction.

DOE stated that capillary diversion of liquid water around large openings is the dominant seepage process providing barrier capability during the ambient period (SAR Section 2.1.2.1). In the context of seepage, DOE defined the thermal period as the time when the drift wall temperature exceeds 100 °C [212 °F]. Thus, DOE used the ambient seepage model to estimate water flux entering the drift from approximately the first few thousand years through a million years (SAR Section 2.3.3.1).

For ambient seepage, DOE described (i) the theoretical treatment of seepage into circular openings in porous media; (ii) its choice of a fracture-only, stochastic, porous-media continuum seepage model; and (iii) field tests at Yucca Mountain used to calibrate seepage models (SAR Section 2.3.3.1). Because drifts are approximately circular in cross section, DOE drew on the theoretical treatment derived from an analysis in Philip, et al. (1989aa) of water diversion around large circular openings in homogeneous porous media. Because water is more likely to drip from fractures than from matrix, and to simplify the numerical models, DOE developed seepage models that include only the fracture network as the porous media. DOE described field tests at Yucca Mountain and observations from analog sites that illustrated the capability to divert water.

DOE implemented a seepage approach predicated on continuum models on the basis of the Richards equation for granular porous media and the representative element volume assumption. Flow in fractures and through the fracture networks, however, may not satisfy these assumptions. For example, the density of fractures with flowing water is small relative to the grid size of the seepage model. Also, in addition to capillary-based flow in small-aperture fractures, flow regimes in fractures also include adsorbed films, sliding drops, rivulet flow, stable thick films, and unstable (laminar or turbulent) films (Ghezzehei, 2004aa); none of these would follow classical Richards-equation-based models. Instead of directly modeling these complexities, DOE's approach calibrated the seepage model to field injection tests that would inherently incorporate small-scale processes and flow regimes in fractures. DOE models solve the Richards equation (Richards, 1931aa) for saturated-unsaturated flow through porous materials, with the van Genuchten–Mualem relations describing the capillary pressure and relative liquid permeability in the fracture continuum as a function of liquid saturation (van Genuchten, 1980aa). Because DOE showed that flow below the PTn can be approximated as steady state (evaluated by NRC staff in TER Section 2.2.1.3.6.3.2) and because field injection tests were performed, NRC staff believes that hysteresis is not an important process to incorporate into the model.

DOE used two separate numerical seepage models: one for calibration to field tests and the other to generate ambient seepage abstraction lookup tables for the performance assessment model. Injection tests at Yucca Mountain, as described in BSC Section 6.2 (2004av), form the basis of the DOE calibrated seepage model that was designed specifically for the injection test domains (SAR Section 2.3.3.2.3.3). The key parameter for estimating seepage is the unsaturated zone property of capillary strength. It is the inverse of the van Genuchten α term (van Genuchten, 1980aa) and reflects the ability of the fracture continuum to offset gravity for water dripping into drifts. DOE conceptually separated percolating water reaching the drift ceiling into (i) water diverted by capillarity, which remains in the host rock; (ii) water dripping from ceiling, which is defined as the seepage flux; and (iii) water entering the drift but not dripping, which includes along-wall flow and evaporation. Because the capillary strength parameter is calibrated from the injection tests, it implicitly includes the effects of evaporation, along-wall flow, and capillary diversion.

The seepage abstraction was developed using the second seepage model that kept the same grid characteristics as the seepage calibration model, but was designed for the geometry of emplacement drifts (SAR Section 2.3.3.2.3.4). DOE used the second seepage model to generate two tables: one for intact drifts and one for collapsed drifts for the seismic modeling cases. These tables covered a wide range of percolation rates, and permeability and capillary strength parameter values. To estimate seepage at any location using the abstraction, DOE sampled capillary strength and permeability from uncertainty distributions (SAR Section 2.3.3.2.4.1) and used a spatially dependent local percolation rate.

DOE adjusted the local percolation flux input for the seepage lookup table by a flow-focusing factor (SAR Section 2.3.5.4.1). Flow-focusing factors increase local percolation in some areas and decrease it in other areas, but the flow-focusing factor does not modify the total flux over the entire area. DOE used the flow-focusing factor to incorporate intermediate-scale heterogeneity (e.g., nonvertical small faults) that might lead to convergence or divergence of flow in the rock layers immediately above drifts. Spatially variable net infiltration and other large-scale heterogeneities from the ambient site-scale unsaturated flow model were propagated to the Multiscale Thermal-Hydrologic Model, and thus were automatically brought into the seepage abstraction. DOE described intermediate scale as falling between the grid scale of the ambient site-scale unsaturated flow model {approximately on the order of 100 m [330 ft]} and several drift diameters. For spatial variability below the scale of several drift diameters, DOE incorporated heterogeneity directly into the seepage numerical model input.

For the seismic ground motion, seismic fault displacement, and igneous intrusion modeling cases, DOE predicted changes to the drift opening that might lead to changes in seepage rate and distribution. To account for changes in the drift wall caused by seismic events that lead to changes in dripping, DOE utilized the second seepage table for collapsed drifts. The degree of drift degradation controls the switch from the intact to the collapsed seepage table. For lithophysal rocks only, values from both tables are obtained and some intermediate value is calculated on the basis of scaling to the volume of rubble detached from the drift ceiling. For nonlithophysal rocks, accumulated rockfall above a specified threshold causes the seepage to be set equal to the percolation rate. The two seismic modeling cases are treated slightly differently. For the seismic ground motion modeling case, all drifts are shifted to a degraded state. For the seismic fault displacement, only a small number of drifts and waste package sections are affected by the seismic event. As with the seismic scenario for nonlithophysal rocks, the DOE seepage abstraction for the igneous scenario is simplified by neglecting the effect of capillary diversion.

NRC staff compared DOE’s description of ambient seepage processes, and the incorporation of those features and processes into models and abstractions, with NRC staff’s understanding of seepage-related features and processes at Yucca Mountain obtained from field observations and independent analyses (NRC, 2005aa; Leslie, et al., 2007aa; Basagaoglu, et al., 2007aa; Or, et al., 2005aa). On the basis of this understanding, NRC staff notes that DOE adequately described features and processes at Yucca Mountain and that DOE included the important features and processes in its conceptual and numerical models for ambient seepage for the nominal, seismic, and igneous intrusion scenarios.

Capillary Diversion around Intact Drifts

DOE described the effectiveness of the unsaturated zone in the Upper Natural Barrier in terms of two metrics: seepage flux and seepage fraction (SAR Section 2.1.2.1.6). This subsection focuses on the seepage flux, which is calculated as the amount of water percolating through the host rock above the drift that the capillary diversion process does not divert around the drift by capillary diversion process. Capillary strength is the key parameter in the DOE seepage model for estimating the amount of water diverted around drifts, both because of the sensitivity of seepage estimates to this input parameter and because of the uncertainty in estimating representative values of this parameter. Percolating water (i) drips from the drift ceiling (seepage); (ii) flows laterally around the drift in the host rock; (iii) enters the drift, but flows along the drift wall; or (iv) enters the drift in the gas phase (vapor flux). DOE defined seepage as only the water dripping from the drift ceiling.

NRC staff reviewed the information provided in the SAR to evaluate the adequacy of DOE's estimate of seepage during the ambient period considering data and model support. NRC staff relied on metrics based on consequence to performance in its evaluation of DOE information provided for the capability of capillary diversion to reduce the flux of water entering drifts. This review approach was driven by NRC staff's difficulty in assessing the effect of uncertainties in DOE's model, due to (i) alternative interpretations of injection tests used to calibrate the seepage model, as described in DOE Enclosure 4 (2009ct) and Or, et al. (2005aa); (ii) representativeness of the locations of injection tests for estimating distribution of capillary strength parameter, as described in DOE Enclosures 2 and 3 (2009ct); (iii) lack of literature support for capillary strength parameter because of grid dependency and nonstandard inclusion of other processes in this term; (iv) alternative conceptual model for water entering drifts suggested by observations in the East-West Cross Drift Passive Test, as shown by Salve and Kneafsey (2005aa) and DOE Enclosure 1 (2009bo); and (v) lack of quantitative support from analogs, and problems and inconsistencies from other analogs [e.g., lithophysae as outlined in DOE Enclosure 6 (2009ct) and ancient artifact sites].

To evaluate seepage rates for performance assessment, NRC staff notes that the 10,000-year and million-year periods may be considered separately. NRC staff evaluates seepage rates for the million-year period in its review of the seepage for degraded drifts in the next subsection. Average seepage estimates during the first 10,000 years are not significantly affected by degraded drifts, because of the low probability of occurrence of seismic and igneous events that lead to collapsed drifts.

For the first 10,000 years after closure, NRC staff notes that underestimates of seepage are not important to 10,000-year performance. This is based on the NRC staff evaluations in TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that note corrosion and mechanical processes do not degrade drip shields during the first 12,000 years after repository closure. Intact drip shields divert all seepage from contacting waste packages.

Capillary Diversion for Degraded Drifts

NRC staff reviewed the DOE approach and estimate of seepage flux that account for the disruptive modeling cases of seismic ground motion, seismic fault displacement, and igneous intrusion. In the DOE model, capillary diversion remains the predominant barrier for seismically degraded drifts. This evaluation focuses on seepage in lithophysal units reflecting seismically degraded drifts.

To address drift collapse, DOE developed a collapsed drift seepage table similar to the intact drift seepage lookup table. The table was developed using an enlarged drift opening, with an 11-m [36-ft] instead of 5.5-m [18-ft] diameter. DOE selected a perfectly circular 11-m [36-ft]-diameter drift opening, as outlined in BSC Section 6.6.3 (2004be), based on inspection of simulation results from rock mechanics modeling, as described in BSC Appendix R (2004al). NRC staff notes that DOE's calculated increase in seepage for the collapsed drift seepage table is solely due to the larger opening and does not reflect irregular drift shapes, as illustrated in BSC Appendix R (2004al). NRC staff believes that both the size and shape of a drift affect capillary diversion. The irregular shape of degraded drifts may lead to increased values of seepage. NRC staff evaluations that follow indicate that uncertainty of seepage due to the irregular shape of collapsed drifts is not significant to performance.

A tiered abstraction was used to account for the degree of drift degradation (SAR Section 2.3.3.4.1.1). For nonlithophysal rock, seepage estimates from the intact seepage table were used with estimated accumulated rubble less than 0.5 m^3 per meter of drift length [5.4 ft^3 per foot of drift length]. Otherwise, seepage was set to the percolation rate, including adjustments from the sampled focusing factor. For lithophysal rock, the intact seepage table was used for accumulated rubble less than 5 m^3 per meter of drift length [54 ft^3 per foot of drift length]. For accumulated rubble greater than 60 m^3 per meter of drift length [650 ft^3 per foot of drift length], the collapsed drift seepage table was used. Seepage was interpolated from the entries in both the intact and collapsed drift seepage tables for intermediate values of rubble accumulation {between 5 and 60 m^3 per meter of drift length [54 and 650 ft^3 per foot of drift length]} in lithophysal rock.

NRC staff notes that DOE estimates of seepage for seismic ground motion, seismic fault displacement, and igneous intrusion modeling cases during the first 10,000 years are reasonable for performance assessment calculations, because of low probability of occurrence for disruptive events. For the seismic fault displacement modeling case, only a small number of waste packages can be affected by each event, as outlined in SNL Section 6.1.2.3.4 (2008ag). Therefore, uncertainty in seepage rates has little effect on total estimated dose. For the igneous intrusion modeling case and for seismically degraded drifts in nonlithophysal rock, seepage is not likely to be underestimated, because DOE conservatively sets seepage rate equal to percolation rate.

For the million-year period, DOE's model determined disruption by seismic ground motion to be the most important contributor to dose estimates. To evaluate DOE's estimates of seepage rate for the million-year period, NRC staff considered the following:

- DOE calculated an average seepage rate for the million-year period to be 49 percent of percolation rate, shown in DOE Table 8 (2010ai). On average, seepage is unlikely to be greater than percolation (i.e., NRC staff knows of no process that focuses water to drifts on an average basis for the repository). Hence, doubling the seepage rate so that seepage is equal to percolation is a maximum bound. Information from sensitivity analyses in DOE Enclosure 1 (2009cx) illustrated that increases in seepage flux by factors slightly larger than two do not significantly affect estimates of total dose.

- The NRC staff expects that degraded drifts provide some measure of barrier capability for seepage, though the percentage of percolation is uncertain. This expectation is based on NRC staff's understanding of the physics of flow at sharp boundaries and modeling efforts completed during the prelicensing period (Basagaoglu, et al., 2007aa; Leslie, et al., 2007aa; Or, et al., 2005aa). The NRC staff models generate seepage rates that are approximately the same as the DOE estimates for the million-year period, thereby providing NRC staff confidence in DOE estimates of seepage.

Therefore, DOE's treatment of capillary diversion and estimates of seepage for the million-year period are reasonable for performance assessment.

Seepage Fraction

DOE described the effectiveness of the unsaturated zone in the Upper Natural Barrier in terms of two metrics: seepage flux and seepage fraction (SAR Section 2.1.2.1.6). This subsection focuses on the latter. DOE defined seepage fraction as the number of drift segments where seepage occurs, divided by the total number of drift segments (SAR Section 2.3.3.1). DOE defined drift segment as an approximation to the average waste package length including the gap between waste packages. Therefore, seepage fraction is essentially the same as fraction of waste packages getting wet. The NRC staff's evaluation of the DOE estimates of seepage fraction focuses on the adequacy of average values used for the million-year period when the seismic ground motion and igneous intrusion modeling cases dominate estimates of dose. Estimates of seepage fraction for the first 10,000 years after closure are not important, because drip shields are expected to remain intact for this entire period (TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) and will divert all seeping water away from waste packages.

Conceptually, DOE defined seepage fraction as the area portion of the repository where dripping is expected to occur, using the footprint of a waste package (drift diameter and waste package length) to define a seepage area. Thus, seepage fraction is linked to the number of waste packages that would get wet if no drip shields were present, divided by the total number of waste packages. The remainder of the repository is the nonseeping environment, where the flux of liquid water potentially dripping in a waste package location is set to zero in the DOE abstraction.

In the DOE abstraction, seepage fraction is important because releases of radionuclides in the seeping environment are transported by advection. In that portion of the repository where the liquid flux is zero, any released radionuclides are transported by diffusive processes out of the waste package, which are slow compared to advective transport rates. Releases in the nonseeping environment rely on transport by diffusion along stagnant water films. Therefore, determination of the threshold at which seepage occurs can impact radionuclide transport.

DOE used the seepage model to predict seepage at all locations. At locations where calculated seepage was less than some small rate, DOE set the value to zero in the performance assessment. Because the seepage fraction is sensitive to the selection of a value for the seepage threshold, DOE Enclosure 5 (2009ct) described sensitivity analyses showing that reducing the seepage threshold value to zero led to a negligible change in performance.

In the abstraction, seepage fraction is fixed at a constant value for any particular TSPA realization. To determine the constant value, DOE selected the highest calculated seepage fraction that would occur at any time during the simulation period (excluding effect of igneous events). This value of seepage fraction was then applied throughout the simulation. Separate TSPA realizations were run for the 10,000-year (using a 20,000-year simulation period) and million-year calculations. DOE provided average values for TSPA realizations in SAR Tables 2.1-6 through 2.1-9. DOE estimated an average seepage fraction of 0.10 for the first 10,000 years when seismic and igneous scenarios do not influence seepage. Similarly, DOE estimated an average seepage fraction of 0.69 for the million-year period when the seismic ground modeling case is the dominant dose contributor. Igneous intrusion and seismic fault displacement scenarios do not influence the seepage fraction used in the million-year calculation, because (i) the abstraction for igneous intrusion sets seepage fraction to one after the occurrence of an igneous event on the basis of SAR Section 2.3.11.6.5 and DOE Enclosure 8 (2009ct), although probability for an igneous event is low for any particular realization, and (ii) seismic fault displacement only affects a small number of drifts.

NRC staff notes that seepage fractions are not underestimated on the basis of the following:

- For the 10,000-year period, DOE adequately demonstrated that drip shields are estimated to remain intact well beyond the first 10,000 years (TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6). Thus, no liquid is predicted to reach the waste packages, because drip shields divert all water regardless of whether the drift segment is a seeping or nonseeping environment. Therefore, the average value, and any uncertainty, in the value of seepage fraction are not important for performance during the first 10,000 years.

- For the million-year period, the NRC staff notes that the calculated seepage fractions cannot increase much before reaching the bounding value of one (i.e., all waste packages get wet). Increasing the seepage fraction from 0.69 to the maximum of 1 at most results in a 43 percent increase in dose, which is not significant to performance results when total dose is low.

The NRC staff expects that seepage fraction during the million-year period will likely be less than the bounding case of one (i.e., all waste packages get wet). The potential for a seepage fraction to be less than one is uncertain, but any value less than one provides further confidence in the reasonableness of DOE's results. NRC staff expects seepage fraction to be less than one because

- DOE observations during a natural seepage event in the South Ramp of the Exploratory Studies Facility tunnel, which started in February 2005 and continued for several months, supported a seepage fraction significantly smaller than one. Results of a DOE simulation using its seepage model qualitatively reproduced the seepage fraction deduced from observations in the tunnel (SAR Section 2.3.3.4.3).

- NRC staff's analysis of site features that may reflect the spacing of flowing fracture suggests the seepage fraction is likely less than one (Basagaoglu, et al., 2007aa). NRC staff's analysis suggests that the average seepage fraction is uncertain, but like DOE's estimate, is likely less than one. The features considered in the NRC staff analysis include (i) fracture spacing, including long through-going fractures; (ii) structural features such as highly fractured zones and faults; and (iii) spacing of features with secondary mineralization.

Representation of Spatial Variability

NRC reviewed the representation and propagation of spatial variability across the repository to determine whether DOE's model underestimated seepage. DOE incorporated spatial variability at several levels in developing its seepage results for performance assessment, including

(i) integration of variability from upstream model results, (ii) variability of permeability and capillary strength in the seepage model, (iii) incorporation of a flow-focusing factor, and (iv) abstraction of spatial variability for performance assessment. The key aspect for NRC staff's evaluation of adequacy of spatial variability reflected in the performance assessment is the upscaling of results to five percolation bins for the entire repository.

DOE incorporated spatial variability in the seepage by

- Integrating aspects of spatial variability related to net infiltration and large-scale heterogeneities from the ambient site-scale unsaturated-flow model (and propagated to the Multiscale Thermal-Hydrologic Model) directly into the seepage model through the input of percolation distribution across the repository. This aspect of variability is evaluated in TER Section 2.2.1.3.6.3.2.

- Incorporating spatial variability and uncertainty in permeability and capillary strength directly into the seepage model used to create the seepage lookup tables. This incorporated variability at the scale of several drift diameters. Permeability was stochastically varied across the seepage model grid. Capillary strength was treated as an upscaled parameter for the model domain.

- Incorporating a flow-focusing factor in the seepage abstraction that addressed the possibility of convergence or divergence of flow in the rock layers above drifts. The flow-focusing factor represents intermediate-scale heterogeneity, which DOE described as falling between the grid scale of the ambient site-scale unsaturated flow model {approximately on the order of 100 m [330 ft]} and several drift diameters (seepage model grid). Flow-focusing factors increased local percolation in some areas and decreased it in other areas, but the total flux over the entire area remained constant in the DOE performance assessment. The resulting values of flow-focusing factors reflect spatial variability and range from 0.116 to 5.016, as outlined in SNL Section 6.6.5.2.3 (2007bk). To provide confidence in estimates of the distribution of flow-focusing factors, DOE performed additional modeling exercises using different assumptions for calculating the focusing factor (SAR Section 2.3.3.2.3.7.6). Results of alternative flow-focusing distributions led DOE to use a narrower range for the distribution of flow-focusing factors.

- Using five percolation bins, and thus five seepage histories, to address spatial variability in the performance assessment. The use of average seepage histories for a percolation bin represents an upscaling of spatial variability.

Because DOE upscaled seepage estimates for its performance assessment to five percolation bins, thus losing details of spatial variability, NRC staff considered whether the upscaling was important to performance. If the upscaling of spatial variability does not lead to underestimates of dose, then the representation and supporting basis of detailed spatial variability are not important for performance. DOE used five percolation bins to separate the repository into areas of similar percolation rates. The areas of any one bin are not necessarily contiguous. The binning of percolation rates roughly ensured spatial continuity of flow zones above and below the repository (i.e., high percolation and thus high seepage zones correspond with high flow zones for transport below the repository). DOE Enclosure 2 (2009bo) compared calculated release results using the five percolation flux bins with results using the 3,264 locations. The analysis showed that the two approaches have similar time histories of radionuclide release, but the bin approach tended to estimate larger repository-wide cumulative release of radionuclide mass to the lower unsaturated zone at 10,000 years and 1 million years after closure. Comprehensive model results, however, predicted higher doses at intermediate times during the first 10,000 years because of radionuclides that are solubility limited and have intermediate to high sorption coefficients, as shown in DOE Enclosure 2, Figures 2 and 3 (2009bo). However, higher predicted dose for the comprehensive model compared to results from the percolation bin approach is not significant, because drip shields are estimated to be intact well beyond the first 10,000 years. DOE adequately represented spatial variability of percolation and seepage flux because DOE showed that the upscaled percolation bin approach in the TSPA abstraction does not underestimate cumulative radionuclide release rates compared to estimates derived from representations with detailed spatial variability.

Thermal Seepage

DOE described two important features created by the thermal pulse that affect seepage into drifts: the dryout zone around a drift and a reflux zone at the outer edge of the dryout zone. DOE's abstraction for thermal seepage sets seepage to zero for drift wall temperatures exceeding 100 °C [212 °F]. This temperature threshold for seepage is the focus of NRC staff's evaluation of thermal seepage in the following paragraphs. NRC staff reviewed the description of features and processes incorporated into the conceptual and numerical models that DOE used to develop the seepage threshold of thermal seepage. Considering uncertainty derived from observations used to develop the thermal seepage abstraction, staff focused its evaluation on the impact to performance.

DOE described the predominant seepage barrier capability for the thermal period as the elimination of liquid flux at the drift wall due to the dryout zone. DOE referred to this as the vaporization barrier (SAR Section 2.1.2.1.6.2). Flow diversion due to capillary forces (capillary diversion) remains a relevant process at all temperatures. DOE indicated that this vapor barrier would eliminate liquid water reaching the drift wall at temperatures exceeding 100 °C [212 °F] (SAR Section 2.3.3.4). In the DOE thermohydrological characterization, two-phase flow (liquid and vapor) in the host rock occurs at the outer edge of the dryout zone. Referred to as the reflux zone or heat pipe, because of increased heat transfer, evaporated liquid water rises and condenses in a continuous cycle. This zone of elevated liquid saturation above the dryout zone can serve as a supply of water added to the local percolation that potentially may breach the dryout zone as focused flow in large fractures, possibly reaching the drift wall and dripping into the drift.

DOE separated the thermal evolution into three regimes: dryout, transition, and low temperature (SAR Section 2.3.5.4.1.1.3). DOE asserted that no water enters the drift during the dryout period, and seepage may occur during the transition period and continue into the lower temperature period transitioning to ambient temperature conditions. DOE defined the dryout period as the time when drift wall temperatures are estimated to exceed 100 °C [212 °F]. DOE eliminated seepage into drifts at a threshold value of 100 °C [212 °F] for intact and partially degraded drifts, but no threshold was implemented for fully collapsed drifts in the seismic scenario. Drift wall temperature is provided to the thermal seepage abstraction from the Multiscale Thermal-Hydrologic Model, which incorporates host rock heat transport, dryout, and rewetting (presented in SAR Section 2.3.5.4; reviewed by NRC staff in TER Section 2.2.1.3.6.3.3).

DOE derived the seepage threshold value of 100 °C [212 °F] from process-level thermohydrological modeling exercises to evaluate the possibility of preferential flow breaching the dryout zone under different realistic and bounding flow conditions. DOE described the thermal seepage model as a dual-continuum (matrix and fracture) representation with coupled heat and mass transport. The model necessarily uses a different property set than that used for the fracture-only continuum models for ambient seepage. DOE assumed that hydrologic properties need not incorporate the effect of thermal-mechanical and thermal-chemical processes. This assumption is based on results of DOE's thermal-mechanical and thermal-hydrological-chemical modeling of the heated field tests, which suggest that changes to the flow patterns are smaller than the variability and uncertainty already considered for seepage. In addition, DOE indicated these changes may be transient and likely would disappear with the decay of the thermal pulse. Generally, the modeling exercises included pulses of water applied to a single fracture and assessment of whether the pulse would evaporate before reaching the drift ceiling. Thermal aspects of the numerical model were supported by field and laboratory observations from thermal tests DOE performed. However, hydrological aspects, particularly preferential flow, are difficult to observe or measure in field tests.

NRC staff compared DOE's description of thermohydrological features and processes during the thermal period, which is summarized in the previous paragraphs, with staff's knowledge and expertise (NRC, 2005aa; Green, et al., 2008aa, and references contained therein) gained from observations and modeling of field and laboratory thermal tests. On the basis of this comparison, NRC staff notes that DOE acceptably described features and processes important for thermal seepage and developed a framework to adequately incorporate those features and processes into the thermal seepage model.

DOE stated that the value of 100 °C [212 °F] for the thermal seepage threshold temperature at the drift wall, being several degrees above the ambient boiling point {96.3 °C [205 °F]}, accounts for modeling uncertainties and the possibility of a heat pipe occurring near the drift wall (SAR Section 2.3.3.3.4). To determine whether the seepage threshold adequately accounted for uncertainty, NRC staff considered field test observations reflective of temperatures at which preferential flow may have occurred.

NRC staff noted that observations at thermal field tests could indicate preferential flow occurring in the dryout zone where measured temperatures exceeded 100 °C [212 °F] (Green, et al., 2008aa). Although DOE provided reasonable explanations for the observations, NRC staff believes the difficulty in collecting relevant observations and the uncertainty in interpretations of observations both support a larger uncertainty factor than reflected by the DOE estimate of the seepage threshold temperature value. DOE Enclosure 7 (2009bo) provided information on drip shield integrity during the thermal period and noted that intact drip shields will divert any dripping water away from waste packages. On the basis of DOE's prediction that the drip shields will remain intact and divert all seeping water away from waste packages throughout the thermal period, DOE's treatment of data uncertainty and model support for the 100 °C [212 °F] thermal seepage threshold are reasonable for performance assessment.

Summary

The NRC staff has reviewed the information DOE provided and notes that the performance and treatment of the uncertainty for the seepage rate and fraction are reasonable because they are consistent with the technical justification provided for the model abstractions and the barrier capabilities. In evaluating seepage rates for performance assessment, NRC staff notes that the 10,000-year and million-year periods may be considered separately. In particular

- For the first 10,000 years after closure, mean values of seepage fraction and rate and their uncertainty are not important for performance assessment, because drip shields are predicted to remain intact well beyond 10,000 years. Intact drip shields divert seeping water away from waste packages.

- The thermal seepage abstraction that shows no seepage occurring when the drift wall temperature exceeds 100 °C [212 °F] is reasonable because drip shields remain intact well beyond the thermal period and thus divert any seeping water away from waste packages.

- For the period from 10,000 years to 1 million years, DOE's seepage and dose calculations are dominated by the seismic ground motion seepage scenario. Average seepage fraction (69 percent) and rate (49 percent of percolation) are reasonable because reasonable increases accounting for uncertainty would not significantly affect performance assessment calculations and dose results.

- Seepage estimates for the igneous intrusion and seismic fault displacement modeling cases are reasonable because DOE uses conservative assumptions for estimating seepage.

2.2.1.3.6.3.5 In-Drift Convection and Moisture Redistribution

This section contains NRC staff evaluation of DOE models, data, and results representing in-drift convection and moisture redistribution. Condensation flux, which results from moisture redistribution via vapor movement, is added to the seepage flux to obtain the total flux of water that may reach the engineered barrier components.

In SAR Section 2.3.5.4.2.1 and SNL Section 6.1 (2007bl), DOE described in-drift convection and moisture redistribution as driven by temperature differences between the waste package, drift wall, and other engineered components. In the DOE conceptual model, decay heat from emplaced waste will create large temperature differences, both radially and axially within a drift. The temperature differences will produce buoyancy-driven natural convective flow of air inside the drift opening that will increase heat transfer and redistribute moisture. Convective air flow will cause water evaporation at warmer locations in the host rock and subsequent transport by in-drift convection to cooler locations where it may condense on cooler surfaces. DOE described the ensemble of evaporation, convective moisture redistribution, and condensation on cooler surfaces inside the drift as the cold trap phenomenon.

The in-drift convection and condensation models provide two outputs. First, the convection model provides support for the effective thermal conductivity used in the thermohydrological model, reviewed in TER Section 2.2.1.3.6.3.3. Second, the condensation model, using dispersion coefficients calculated from the convection model, provides probability and flux of condensation to performance assessment. Condensation is added to the dripping flux to obtain a total seepage flux entering drifts. Condensation is linked to DOE's abstraction for chemistry of liquid water contacting engineered barrier components (reviewed in TER Section 2.2.1.3.3) and to flux of water in the invert, which influences radionuclide transport (reviewed in TER Section 2.2.1.3.4).

The NRC staff evaluation of the convection and condensation models and results are divided into two parts. The following two subsections contain NRC staff's evaluation of (i) in-drift heat transfer and convection and (ii) moisture redistribution and condensation.

2.2.1.3.6.3.5.1 In-Drift Heat Transfer and Convection

NRC staff reviews the DOE conceptual model for in-drift heat transfer and implementation of the numerical model in this section. The review considers the adequacy of the heat transfer model to estimate representative dispersion coefficients and effective thermal conductivity.

In-drift heat exchange processes involve conduction, convection, radiation, and phase-change (latent) heat transfer (SAR Section 2.3.5.4.2.3.1). Heat transfer processes reduce temperature differences created by emplacing heated waste packages in a drift (SAR Section 2.3.5.4.2.3). Though the heat transfer model generally will be referred to here as the convection model, radiative and conductive heat transfer processes are also included in DOE's model and analyses.

DOE implemented the convection model using the commercial computational fluid dynamics solver FLUENT®, a code commonly used in industry and academia. DOE set up FLUENT® to solve the steady-state form of the Navier-Stokes equation for selected times during the thermal pulse with a model domain that is divided into a large number of computational cells. DOE models incorporate, as appropriate, the complex arrangement of engineered components and take advantage of vertical axial symmetry to reduce computational effort. Radiation and conduction are included, but latent heat transfer is excluded because DOE showed in SNL Sections 6.3.7.2.4 and 6.3.5.1.2 (2007bl) that it does not significantly affect overall heat transfer and convection. Independent NRC experiments and analyses (Fedors, et al., 2004aa) similarly show latent heat transfer is a small component of overall heat transfer. For radiation and conduction, DOE used standard textbook heat transfer models and relevant parameter values (e.g., Incropera and DeWitt, 2002aa; Kreith and Bohn, 2001aa).

NRC staff focused its review of the heat transfer on the convection aspect of the DOE model because convection most directly affects moisture redistribution and is the most complex of the heat transfer processes in the drifts. Asymmetric geometry inside the drift leads to complex convective flow patterns at different scales. SNL Sections 6.1 and 6.2 (2007bl) used numerical models at local and drift scales to represent heat transfer at scales ranging from large-scale heat transfer along drifts (center to repository edge) to small-scale heat transfer across boundary layers at solid-air interfaces. The DOE local-scale model emphasizes cross-sectional patterns in its simulations of temperature gradients between the waste package, drip shield, and drift wall. The DOE drift scale models address temperature gradients between the hot repository center and cooler edges. Model support was provided by DOE laboratory convection experiments and other experiments in the general literature using similar geometries (Kuehn and Goldstein, 1978aa).

NRC staff notes that DOE has adequately incorporated convective patterns in its conceptual and numerical models for convection. NRC staff notes that analysis at two different scales is a generally accepted scientific technique for simulating large systems covering a wide range of scales and incorporating complex geometries. Furthermore, NRC staff notes that DOE adequately described and incorporated the important heat transfer processes and engineering design features in developing its conceptual and numerical convection models. Independent NRC experiments and analytic modeling results (Das, et al., 2007aa; Green and Manepally, 2006aa; Manepally, et al., 2007ab) are consistent with DOE's assessments.

In its two- and three-dimensional convection models, DOE used dimensions and physical properties of waste packages, drip shield, invert, and heat loads consistent with upstream models, as shown in SAR Sections 2.3.5.4.2.2 and 1.3.2 and SNL Section 4.1 (2007bl), and used values for physical properties of fluids and solids derived from standard thermal textbooks (i.e., Incropera and DeWitt, 1996aa; Kreith and Bohn, 2001aa). DOE assumed that convection is based on pure air (i.e., without water vapor) and showed that this assumption would slightly underestimate in-drift vapor transport, as outlined in SNL Section 6.1.3.2.1 (2007bl). Independent NRC staff analyses similarly used and justified this assumption (Fedors, et al., 2004aa). On the basis of this assumption, DOE used a neutrally buoyant tracer gas in the simulations and calculated the dispersion coefficients using the resulting concentration gradients. Independent analyses reported in Fedors, et al. (2004aa) support use of this assumption. On the basis of these considerations, NRC staff notes that DOE adequately represented physical parameters for solids and fluids and geometrical parameters in convection flow analyses.

NRC staff considered DOE's use of the convection model results to estimate dispersion coefficients for the condensation model. DOE stated that dispersion coefficients are dependent on a number of factors, including axial drift wall temperature variation, convective flow pattern, presence of drip shields, and time, as outlined in SNL Section 6.2.7 (2007bl). DOE calculated dispersion coefficients at two locations in the simulated drift and at discrete timesteps during the thermal pulse. DOE addressed uncertainty in the dispersion coefficient using parametric studies and bounding analyses, as described in SNL Sections 6.1.7 and 6.2.7 (2007bl). NRC staff reviewed the representativeness and uncertainty of the calculated dispersion coefficients for their intended usage with the condensation model. In particular, DOE did not address some uncertainty in the convection model output, including (i) the basis for the representative dispersion coefficients being spatially constant, rather than varying along drifts; (ii) justification for selecting the two specific locations in the analog 71-m [233-ft] drift for calculating dispersion coefficients and how the values are representative of an entire drift length; and (iii) the effect of the revised heat load scenario (SNL, 2008ai) on calculations. Although the DOE convection model is reasonable for calculating dispersion coefficients, the approach for estimating representative values from the model output may not be fully supported for use in the condensation model, because of the three aspects of uncertainty in representative dispersion coefficients described previously. However, NRC staff notes that the choice of dispersion coefficient values is not important for repository performance. This is supported by the NRC staff evaluations noting that drip shields remain intact for 12,000 years (TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) and thus prevent direct contact of condensation with waste packages during the first 2,000 years, when DOE maintains condensate formation is significant.

NRC staff also reviewed DOE's use of the convection model results to derive estimates of effective thermal conductivity for the porous media representation of air gaps in the thermohydrologic submodel of the Multiscale Thermal-Hydrologic Model. DOE computed an effective thermal conductivity from output of the convection model between (i) drip shield and drift wall and (ii) drip shield and waste package, as shown in SNL Table 6.4.7-3 (2007bl). The calculated values supported the Francis, et al. (2003aa) correlations used in the Multiscale Thermal-Hydrologic Model submodels, as described in SNL Appendix I[a] (2007aj). NRC staff notes use of the convection model to estimate effective thermal conductivity is reasonable because it follows a widely accepted, technical approach used in engineering analyses (Kuehn and Goldstein, 1976aa, 1978aa) and the inputs are reasonable for the purpose of the model.

In summary, DOE provided reasonable support for estimates of in-drift heat transfer because (i) the convection model is based on a reasonable technical approach, (ii) inputs are supported by textbooks values, (iii) data and model uncertainty were addressed with parametric studies and bounding analyses, and (iv) model inputs and results are supported by laboratory experiments. NRC staff also notes the approach DOE used to estimate thermal conductivity is reasonable, but the support for dispersion coefficients relied on drip shield integrity.

2.2.1.3.6.3.5.2 Moisture Redistribution and Condensation

NRC staff reviewed the information DOE presented to support estimates of condensation flux in drifts. DOE described the conceptual, numerical, and abstraction models for moisture transport and condensation in SAR Section 2.3.5.4.2. Treatment of data and model uncertainty was described in SAR Sections 2.3.5.4.2.2 and 2.3.5.4.2.3.3. Considering the support DOE provided for models and results, NRC focused its evaluation on the consequence of condensation flux on repository performance.

Condensation Approach

This section reviews the DOE description of the conceptual, numerical, and abstracted models used to estimate condensate rate for the performance assessment model.

Moisture redistribution and condensation inside the drift is also referred to as the cold trap process. The process involves water evaporation from hotter locations, convection to cooler locations, and the condensation of vapor on cooler surfaces. DOE considered surface condensation, which requires direct contact of the convecting gas-phase with a cooler surface, but did not provide information on the potential effects of dust or volumetric condensation (Cussler, 1995aa) in its conceptualization. DOE predicted that condensation will only occur on the drift wall because the drift wall will be cooler than the drip shield, waste package, or invert at each axial position along any drift. DOE added condensation flux to the dripping rate to obtain a total seepage rate contacting engineered barrier components and reaching the invert and, thus, affecting advective radionuclide transport rates to the natural system.

DOE described the evolution of moisture transport and condensate formation using three stages controlled by drift wall temperature (SAR Section 2.3.5.4.2.1). In Stage 1, the initial cooling stage, the drift wall temperature exceeds boiling along the entire length of the emplacement drift. DOE stated that no condensate formation takes place during the initial stage. In Stage 2, the intermediate cooling stage, the drift wall temperature exceeds boiling in most of the drift, but the end of the drift (repository edge) is below the boiling temperature. For the intermediate stage, DOE performed a bounding analysis to calculate condensation flux occurring on codisposal waste packages at cooler locations. In Stage 3, the final cooling stage, the drift wall temperature is below boiling along the entire length of the drift. In the DOE abstraction, condensation occurs at both codisposal and spent nuclear fuel waste package locations, but all condensation ceases at 2,000 years. Results for process-level models for the intermediate and final cooling stages provide the basis for the abstraction model used in the performance assessment.

For the intermediate cooling stage, DOE estimated condensation using a three-dimensional, pillar-scale thermal-hydrological model (SAR Section 2.3.5.4.2.4). This is an alternative conceptual model supporting the thermohydrological results the Multiscale Thermal-Hydrologic Model calculated (SAR Section 2.3.5.4.1.3.3; reviewed in TER Section 2.2.1.3.6.3.3). Described as a bounding approach, DOE used a range of dispersion coefficients and percolation values in the three-dimensional, pillar-scale model to determine that condensation occurs on codisposal waste packages, but not on spent nuclear fuel waste packages.

For the final cooling stage, DOE used a one-dimensional analytical moisture transfer model to estimate condensation occurrence and flux when drift wall temperatures along the entire length of the drift are below boiling. The DOE network model calculates quantity of condensate at a given location along the drift (SAR Section 2.3.5.4.2.3.1) for specified percolation rates and thermal input. The one-dimensional model is based on a diffusion-type equation and uses values of dispersion coefficients calculated by the convection model as an effective diffusion-type parameter. Conductive heat transfer in host rock is based on an analytical mountain-scale conduction model, as outlined in SNL Section 6.3.5.1.1 (2007bl), and in-drift heat transfer between components is calculated on the basis of correlations derived from simple systems and reported in open literature (Raithby and Hollands, 1985aa; Kuehn and Goldstein, 1976aa; Burmeister, 1993aa). DOE considered the supply of water for evaporation at drift walls to be bounded by the sum of capillary-pumping flow and local percolation flux intercepted by the emplacement drift footprint. The NRC staff considers the control parameter that commits DOE to an unheated open length at the ends of emplacement drifts (SAR Section 1.9, Control Parameter Number 01-18, Table 1.9-9) to be an important design feature that was integrated into the convection and condensation models. This design feature allows axial convection to convey a portion of the moisture beyond the last waste package before condensation would occur.

DOE implements the abstraction of condensation in the performance assessment using a three-step process (SAR Section 2.3.5.4.2.4). First, DOE used the process-level condensation models to generate a set of results for different parametric variations that account for dispersion coefficient, percolation rates, invert assumptions, and temporal variation of heat load. Second, DOE developed a set of regression curves that establishes a functional relationship between percolation flux, probability of condensation, and condensate mass. Third, DOE used the regression curves for each percolation subregion to determine the occurrence (fraction of area) and magnitude of condensation. DOE added condensate flux directly to dripping flux to obtain a total flux of water entering drifts.

In DOE’s model, condensation within emplacement drifts would be altered if a disruptive event occurs during the thermal period. The DOE abstraction sets condensation to zero once an igneous intrusion or drift collapse event occurs (SAR Section 2.4.2.3.2.1.12.3), because these processes would fill drifts with rock and substantially reduce air gaps. However, such events have low probability and DOE expects drifts to remain intact throughout the first 2,000 years, as described in DOE Enclosure 7 (2009ct).

DOE provided a transparent description of conceptual, numerical, and abstracted models and parameter inputs. This is supported by staff expertise gained from prelicensing interactions with DOE (NRC, 2005aa) and from independent laboratory experiments and numerical modeling of convection, vapor transport, and condensation in drift analogs (Das, et al., 2007aa; Manepally, et al., 2007ab).

Condensation Results

The DOE condensation model results can be summarized as follows:

- During the intermediate cooling period, all codisposal waste package locations receive condensation dripping from the drift ceiling at rates 8 to 35 times greater than the mean seepage rate [calculated from SAR Table 2.1-11 and DOE Table 7 (2010ai)]. DOE conservatively applied condensation to all codisposal waste package locations, but no spent fuel waste package locations receive any condensation.

- During the final cooling stage (after approximately 1,500 years), mean condensation rates are less than 1 percent of mean seepage rate, and condensation only occurs at a small fraction of locations for both codisposal and spent fuel waste packages, as shown in SAR Tables 2.1-10 and 2.1-11 and DOE Tables 6 and 7 (2010ai).

The average condensation rate in a percolation bin is calculated by multiplying the fraction of waste package locations receiving condensation times the condensation flux rate, which is then added to the seepage rate (SAR Section 2.3.3) rate to obtain a total dripping rate.

NRC staff considered two types of information DOE provided to gain confidence in the condensation model and results used in the performance assessment. First, in developing the conceptual model, DOE stated that observations of vapor movement and condensation in response to small thermal gradients in the East-West Cross-Drift indicate the importance of the cold trap process in the repository (SAR Section 2.3.5.4.2). Although no quantitative estimate could be made from observations in the East-West Cross Drift, the NRC staff believes the observations from near-ambient conditions point to possible condensation in emplacement drifts both prior to and well beyond the 2,000-year cutoff used in the DOE condensation abstraction. Second, to provide confidence in model results, DOE included evaluations by two university professors in the model validation section of SNL Section 7.6 (2007bl).

On the basis of its evaluation of condensation estimates on the consequence for repository performance, the NRC staff notes that condensation estimates are not important for repository performance. This is derived from different considerations for different time periods following repository closure. During the first 2,000 years after permanent closure, the presence of the intact drip shield ensures that the condensate will not directly contact waste packages. During this period, DOE asserts that drip shields will be sufficiently warm that any condensation will occur on drift walls, above or away from drip shields. On the basis of the evaluation of drip shield corrosion and mechanical degradation processes, NRC staff notes in TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that drip shields are likely to remain intact for at least 12,000 years, which is well beyond the thermal period DOE defined for condensation.

NRC staff notes that even if condensation continued past the 2,000-year limit DOE specified in the TSPA, it will not significantly affect repository performance for the following reasons:

- For condensation during the period from 2,000 years to 12,000 years after closure, condensation rates are not important, because the drip shields will remain intact and thus divert water from the waste package. Note that the DOE condensation rate estimate is zero during this period, though NRC staff considers the possibility that evaporation and condensation rates are nonzero based on observations in the East-West Cross-Drift Test.

- For condensation between 12,000 years and 255,000 years after closure, NRC staff expects the total amount of water entering drifts would be the same regardless of the mechanism for water entering drifts. The total flux of water approaching a drift limits the total water entering a drift regardless of whether that water flux is due to dripping or to evaporation and condensation. NRC staff notes in TER Section 2.2.1.3.6.3.4.1 that the seepage percentage is already high for this period and that further increases would minimally affect performance. In addition, NRC staff suggests that any change in the fraction of waste package locations getting wet because of condensation, rather than dripping, will not adversely affect performance, because DOE already set the seepage fraction to be a high value (TER Section 2.2.1.3.6.3.4.1) for this period.

- Beyond 255,000 years, DOE predicts that drifts will be collapsed (SAR Section 2.1.2.2.6) and axial convection along drifts will no longer occur. If portions of the drifts collapsed, allowing for small convection cells and potential condensation, NRC staff expects the fraction of waste packages that get wet to remain the same or decrease.

Summary

On the basis of its consideration of DOE-provided information on in-drift convection, moisture transport, and condensation abstraction, the NRC staff notes that DOE acceptably described and represented convection along drifts to estimate dispersion coefficients and support estimates of effective thermal conductivity in the Multiscale Thermal-Hydrologic Model. The NRC staff notes that, for the condensation abstraction and results, condensate estimates are not important for performance during the first 2,000 years, because drip shields are expected to remain intact. NRC staff notes in TER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that drip shields are expected to remain intact and divert water away from waste packages for 12,000 years. Drip shield performance is evaluated in TER Sections 2.2.1.3.1 and 2.2.1.3.2. After the drip shield becomes degraded, NRC notes that condensation flux estimates would not change the total flux of water entering drifts, because dripping rates would decrease in response.

2.2.1.3.6.3.6 Ambient Mountain-Scale Flow—Below the Repository

NRC staff's evaluation of the flow field in the unsaturated zone below the repository considers how the flow magnitudes and patterns affect radionuclide transport. Flow above the repository is evaluated in TER Section 2.2.1.3.6.3.2, and flow below the repository is evaluated in this section. The site-scale unsaturated flow model provides flow fields both above and below the repository for different climates (SAR Section 2.3.2.3). In TER Section 2.2.1.3.6.3.2, NRC staff evaluates the use of site characterization data, development of a conceptual model, calibration procedure, and confidence building exercise and validation for this model. The following factors influence aspects of the ambient site-scale flow fields that are relevant to the flow fields below the repository: (i) the CHn influences flow in the southern and northern portions of the repository footprint, (ii) the active fracture model (AFM) influences the fracture-to-matrix flux, and (iii) the uncertainty of flow fields influences transport. Output of the ambient site-scale flow model (i.e., flow patterns, water saturations, and flow rates) is direct input to the radionuclide transport abstraction, which NRC staff reviews in TER Section 2.2.1.3.7.

Evaluation of the adequacy of flow fields below the repository is separated into three parts: (i) NRC staff reviews the DOE description of the conceptual model for flow below the repository, (ii) NRC staff evaluates information and observations supporting flow features in the southern vitric CHn zone and in the northern zeolitic CHn zone, and (iii) NRC staff evaluates how uncertainty in flow fields can affect repository performance regarding radionuclide transport.

2.2.1.3.6.3.6.1 Flow Model Conceptualization

DOE described aspects of the flow below the repository in SAR Section 2.3.2.2.1.4 and how these aspects are related to the hydrogeologic units (SAR Table 2.3.2-2) below the repository. The hydrogeologic units include

- TSw (Topopah Spring welded tuff); dominantly fracture flow

- CHn (Calico Hills nonwelded hydrologic units);

  - Calico Hills Formation; nonwelded vitric and zeolitic zones
  - Prow Pass Tuff and top of Bullfrog Tuff; devitrified and zeolitic horizons

- CFu (Crater Flats undifferentiated units); varied degree of welding

- Fault zones crossing all hydrologic units

DOE described flow in the first layer underlying the repository, the TSw, as occurring dominantly through fractures (SAR Section 2.3.2.2.1.3). DOE assumed steady-state flow was based on dampening of episodic infiltration pulses by the overlying PTn unit (SAR Section 2.3.2.2.1.2; reviewed by NRC staff in TER Section 2.2.1.3.6.3.2). Percolating water moves approximately vertically from the ground surface through the proposed repository to the base of the TSw. Below the TSw, DOE described flow patterns in the CHn that differ markedly between the northern and southern portions of the repository footprint (SAR Section 2.3.2.2.1.4). The CHn is the only unit where lateral variation has been incorporated into the ambient site-scale unsaturated flow model. DOE described portions of the originally vitric CHn layer as altered to zeolites, which strongly modifies hydraulic properties. The distribution of alteration is described as increasing with depth and increasing to the north and east across the repository footprint (SAR Section 2.3.2.2.1.4). Below the southern portion, DOE expects flow in the vitric CHn to be dominated by matrix flow, because matrix permeability is higher than percolation rates. DOE expects little fracture flow where the CHn is unaltered. Below the northern portion, where most of the CHn has been altered to zeolites, perched water occurs in overlying units due to low permeability of the zeolitic tuff. DOE described that perched water affected performance by causing lateral flow to faults and fast vertical flow and transport down to the groundwater table (saturated zone). Because flow through the matrix of vitric CHn units is much slower than flow through fractures and faults, DOE predicted travel times in the southern portion to be much longer than in the northern portion of the repository (SAR Section 2.3.8.1).

Little information was available to support DOE model estimates of flow patterns in the underlying Prow Pass, Bullfrog, and CFu Tuffs. DOE described these units as layers of devitrified, zeolitic, welded, and nonwelded tuff (SAR Table 2.3.2-2). In SAR Section 2.3.2.2.1.4, DOE noted that these units comprise a small volume of rock above the water table. With contrasting hydraulic properties for layers in Prow Pass, Bullfrog, and CFu Tuffs, percolating waters would be expected to switch between fracture- and matrix-dominated flow several times between the repository horizon and the water table, with the matrix layers controlling sorption and travel time. NRC staff notes that DOE adequately described the flow magnitudes and paths below the repository consistent with the staff analysis (Leslie, et al., 2007aa; NRC, 2005aa).

2.2.1.3.6.3.6.2 Flow Features Below Southern and Northern Portions of Repository

NRC staff reviewed the support DOE provided for flow features below the repository that may affect performance. Flow patterns below the repository can be described in terms of water velocity, which together with water saturation is directly tied to transport travel times. Flow patterns can be separated into three horizons starting at the repository and proceeding downward: (i) fracture flow in welded tuffs of the TSw, (ii) influence of nonwelded tuffs of the CHn, and (iii) flow in the variably welded tuffs below the CHn. DOE described flow through the TSw as primarily vertical and rapid because of the pervasive fracture network, but described rock alteration in the underlying CHn as causing different flow patterns in the southern and northern portions of the repository footprint (SAR Section 2.3.2.2.1.4 and supporting documents). In the southern portion, travel times are slow and sorption potential is high due to flow predominantly through the matrix of the unaltered, vitric CHn. This is contrasted with fast travel times for transport in the northern portion of the repository where fracture and fault flows dominate, because the low permeability of the altered, zeolitic CHn led to the formation of perched water. Below the CHn, alternating layers of tuff with differing degrees of welding are host to the present-day and expected future water table.

There is limited access to, and thus limited direct observations of, these units because of their depth below the ground surface and below the existing tunnel and drift. Therefore, NRC staff considered how uncertainty caused by sparse data may affect performance of the repository. To accomplish this, the NRC staff reviewed support for the conceptual model and estimation of parameter values of the numerical model deemed important for flow patterns below the repository. These aspects include (i) fracture-matrix flow in the TSw immediately below the repository, (ii) properties and distribution of vitric CHn in the southern portion below the repository, (iii) influence of zeolitic CHn on perched water below the northern portion of the repository, and (iv) uncertainty of flow patterns below the CHn.

Flow in Welded Topopah Spring Tuff

NRC staff reviewed the basis provided for flow patterns through the fracture network of the welded tuffs of the TSw immediately below the repository. The review here focuses on support for the hydrologic properties of the fracture network, including support for the conceptualization and estimation of parameter values for the AFM.

DOE utilized air permeability and fracture data (SAR Section 2.3.2.3.3.2) as prior information for calibration of fracture hydrological properties. DOE assumed that fractures follow the van Genuchten–Mualem constitutive relations for saturation, water potential, and relative permeability, adjusted by the AFM. Support for the AFM parameter values is discussed later in this chapter. DOE based fault hydrological properties on air permeability measurements (SAR Section 2.3.2.3.3.3) and integrated these properties into the transition of one-dimensional calibration values to three-dimensional values across the site. DOE did not include sorption on fracture surfaces (SAR Section 2.3.8.1) and represented flow through welded-tuff fractures and faults as fast [e.g., tens to hundreds of years; SAR Figure 2.3.8-49 and DOE Enclosure 6 (2009an)]. The NRC staff notes that DOE represented uncertainty in the fracture and fault input parameters of permeability, water retention, and relative permeability because alternative representations would either insignificantly reduce radionuclide travel times or improve repository performance.

Estimation of parameter values for the AFM may affect performance because the AFM controls the flux of water from fractures into the surrounding matrix. Increasing the flux of water moving from fractures to matrix increases the movement of radionuclides into the matrix, where slower travel times and increased sorption occurs relative to the fracture continuum. Therefore, NRC staff reviewed the basis and uncertainty of parameter values used for the AFM.

DOE described and implemented the AFM of Liu, et al. (1998aa) in the site-scale unsaturated flow model to capture the effects of gravity-driven fingering flow through a limited number of water-conducting or active fractures. In the site-scale unsaturated zone flow model, DOE kept layerwise AFM parameters constant in TSPA realizations, but varied the values with infiltration uncertainty scenario, as shown in SAR Tables 2.3.2-8 through 2.3.2-11 and SNL Section 6.5.6 (2008an). DOE estimated the AFM parameters for the three-dimensional model by calibrating one-dimensional flow simulations with field data (SAR Section 2.3.2.4.1.2.3.2). DOE adjusted the calibrated AFM parameter values to induce perching for model layers with observed perched water, thereby forming a fast pathway for water to flow into faults and bypass the underlying low-permeability and high-sorption units. The DOE AFM model is reasonable for representing flow patterns below the repository because DOE showed it was capable of reproducing field-injection test data and natural geochemical and isotopic observations. Additional confidence in this evaluation is derived from independent NRC staff analysis and modeling studies (Basagaoglu, et al., 2009aa).

NRC evaluated the support DOE provided for representative estimates of the AFM parameter values used in performance assessment calculations. NRC staff considered the (i) uncertainty from using one-dimensional model sensitivity analyses to estimate three-dimensional behavior for a three-dimensional model result and (ii) uncertainty in interpretations of observations from natural isotopes, fracture coatings, and forced-injection field tests caused by multiple processes affecting transport of natural and injected tracers. DOE estimation of AFM parameter values is reasonable because DOE provided results of sensitivity analyses that showed no significant changes to the flow fields for reasonable ranges of the AFM parameter values (SAR Section 2.4.2.3.1.7 and references cited therein).

NRC staff also considered integration between the unsaturated flow and transport for the AFM parameters. DOE used different values and uncertainty distributions for the AFM parameter values for flow and transport simulations. DOE used fixed values of the gamma parameter in the AFM in flow simulations (SAR Tables 2.3.2-8 to 2.3.2-11, 2.3.2-13, and 2.3.2-21 to 2.32-24). DOE, however, treated the same parameter as uncertain in transport simulations by probabilistically sampling from a distribution of values (SAR Section 2.3.8.4.5.2). From a conceptual perspective, DOE, following Zhou, et al. (2007aa), suggested that advective flow occurs mostly in large-scale fractures, but transport through matrix diffusion takes place in both small- and large-scale fracture networks in real heterogeneous fractured continua. Thus, conceptually, there may be additional contributions to the active fracture-matrix interfacial area, across which diffusive mass transfer occurs, by small-scale fracture networks that do not contribute to large-scale flows. The effect of transport in small-scale fractures of a network is not explicitly addressed in the DOE algorithm, as specifically shown in SNL Eq. C–40 (2008an), but the effect is incorporated in the sampling of gamma for transport calculations. NRC staff reviews the distribution of gamma parameter values for transport in TER Section 2.2.1.3.7.3.2.3. Because DOE provided a reasonable basis for using different values for the AFM parameter in the flow model compared to those in the transport model, NRC staff notes the flow and transport models are adequately integrated in regard to the AFM.

Influence of CHn in Southern Portion of Repository

DOE described the CHn in the southern portion of the repository as unaltered, though some zeolitic alteration is present and increases with depth. DOE identified the unaltered, or vitric, zones of the CHn as an important component of the Lower Natural Barrier in terms of water travel times and the unit's capability for delaying radionuclide movement (SAR Section 2.1.2.3.1). Because the travel times through the matrix of the vitric CHn are longer than in other units above and below the CHn where fast fracture flow may dominate, transport through the vitric CHn dominates the travel times of the entire sequence of hydrogeologic units below the southern portion of the repository to the water table. Therefore, NRC staff included the uncertainty of the hydrologic properties and the spatial distribution of the vitric CHn in its review.

DOE characterized flow in the vitric CHn unit as matrix flow dominant (i.e., little or no fracture flow) due to the unit's relatively high matrix permeability and porosity (SAR Section 2.3.2.2.1.4). DOE provided information from boreholes near the repository and from the Busted Butte analog site to support its characterization of the vitric CHn below the repository. Busted Butte was a field experiment DOE performed to support the importance of capillarity and matrix-dominated flow in the CHn vitric tuff using several injection tracer tests (SAR Section 2.3.2.3.2.4). Noting that the CHn at Busted Butte is a distal portion of the CHn found at Yucca Mountain, as described in BSC Appendix H (2004av), DOE provided a lithologic and mineralogic comparison of the CHn near the repository with the units at Busted Butte, but did not provide a hydrologic comparison. Therefore, the NRC staff reviewed measured and calibrated values of hydraulic conductivity and porosity from the boreholes near Yucca Mountain and the Busted Butte site. The NRC staff notes that porosity values do not significantly differ between the two sites. The NRC staff noted, however, that hydraulic conductivity does differ between the sites. The NRC staff analysis indicates uncertainty in hydraulic conductivity is two orders of magnitude larger or smaller than the values used in the performance assessment, when considering measured values in different layers and scale effects for calibrated values. Values used in the NRC staff analysis are in (i) SNL Tables 7-8, 7-9, 7-13, and 7-14 (2007bj); (ii) SAR Table 2.3.2-3; (iii) BSC Table H-3 (2004av); and (iv) Flint Table 7 (1998aa). The NRC staff notes that differences between the borehole information at Yucca Mountain and Busted Butte are not important because

- If hydraulic conductivity of the CHn matrix near the repository footprint is several orders of magnitude smaller than estimated, NRC staff expects matrix flow to dominate in the vitric CHn unit because the matrix can still accommodate typical percolation rates.

- If the hydraulic conductivity of the CHn matrix is larger than estimated by DOE, travel time and sorption would not be significantly different.

Because the uncertainty in the range indicated by the Busted Butte experiment is not important to performance, the NRC staff notes that DOE estimates of hydrologic properties of the vitric CHn are reasonable for performance assessment.

DOE described the spatial distribution of the CHn both laterally and vertically in the southern portion of the repository footprint. DOE indicated there were few available data to constrain the spatial distribution of vitric and zeolitic zones below the repository. As a result, uncertainty about spatial variability of the CHn units (i.e., vitric versus zeolitic) may increase with depth and distance from the repository footprint (SAR Section 2.3.2.3.5.3). In its analysis, DOE incorporated data from surface mapping and 23 boreholes spread in and around Yucca Mountain, as described in BSC Section 6.2.3 (2004bt). Six of these boreholes lie within or near the edge of the repository footprint.

To assess the reasonableness of using a small number of boreholes to represent the extent of the vitric CHn, the NRC staff evaluates the uncertainty of the spatial distribution of the vitric tuff by considering potential alternative representations. NRC staff notes that (i) the contact between the zeolitic and vitric zones of the CHn is not well constrained because of large distances between boreholes, and the horizontal and vertical complexity of the contact itself and (ii) by fixing the distribution of CHn vitric and zeolitic zones in the site-scale unsaturated flow model grid, DOE did not quantify uncertainty in the extent of the vitric unit in performance assessment calculations. The NRC staff built confidence in its review of the spatial distribution of vitric tuff using a bounding estimate for the performance consequence resulting from the presence of vitric tuff. Because of slower flows and greater sorption in matrix-dominated flows through the vitric tuff, which DOE represents as underlying approximately half of the repository, the vitric tuff provides a greater barrier capability than elsewhere. In the alternative representation, the NRC staff assumed that the southern vitric portion performed similarly to the northern zeolitic portion, thereby substantially reducing the performance capability of the vitric portion. If the entire repository was represented by the generally rapid travel times and minimal sorption DOE ascribed to the northern zeolitic half of the repository footprint, then dose would at most increase by a factor of two. With this extreme bounding assumption, alternative vitric distributions would either reduce calculated dose (if the vitric area were larger) or would increase by no more than a factor of two (if no vitric unit was present). This bounding analysis provides confidence that uncertainty in the location of the vitric/zeolitic contact, and thus the spatial distribution of the vitric unit, has small consequence for performance relative to the DOE model. Therefore, using a performance metric, NRC notes that the small number of boreholes available to constrain the distribution of the vitric CHn is reasonable. The NRC staff compared DOE's description, integration, and interpolation of sparse borehole data with staff's knowledge of site characteristics, as outlined in Leslie, et al. Section 6.4 (2007aa) and NRC Section 5.1.3.6.4 (2005aa). The NRC staff notes that DOE reasonably represents the estimated areal extent and vertical variations because DOE represents the contact between vitric and zeolitic units as consistent with geological conceptualizations considering the interplay between faulting and stratigraphic dip of the layers.

Influence of CHn in Northern Portion of Repository

DOE described released radionuclides in the northern portion of the repository as starting out in the welded units of the TSw; proceeding vertically, predominantly in the fracture system to perched water above the zeolitic zones of the CHn; predominantly bypassing the low-permeability, high-sorptivity zeolitic zones by rapidly moving laterally to faults in the perched water body; and finally rapidly moving vertically within faults to the water table (SAR Section 2.3.2.2.1.4). DOE did not include sorption on fracture surfaces (SAR Section 2.3.8.1) and represented flow through welded-tuff fractures and faults as fast [e.g., tens to hundreds of years; SAR Figure 2.3.8-49 and DOE Enclosure 6 (2009an)]. DOE stated that neglecting sorption on fracture surfaces and representing fast flow in fractures and faults were conservative assumptions for estimating dose in the performance assessment. In this section, the NRC staff reviews the DOE treatment of model uncertainty for (i) perching of water and (ii) extent of the zeolitic unit that causes the perching to ensure that the DOE representations did not lead to underestimates of dose.

DOE implemented a permeability-barrier conceptual model for perched water (SAR Section 2.3.2.4.1.2.4.4) in which sufficient local percolation flux, poorly interconnected and conductive fractures, and locally low vertical and horizontal permeabilities contribute to the occurrence of perched water. DOE incorporated the conceptual model for flow in the perched water by adjusting the calibrated model parameters for the layers where perched water has been observed in the field. DOE identified a reasonable alternative model for perching of water whereby slow vertical flow through the zeolitic portions of the CHn unit occurs at rates much smaller than percolation (SAR Section 2.3.2.4.2.1.3). Of the two reasonable perched water models, DOE selected the one that provides the smallest barrier capability. Therefore, the NRC staff notes the permeability-barrier conceptual model adequately represents field conditions for performance assessment.

DOE constrained the spatial extent zeolitic CHn, and thus perched water, in the site-scale unsaturated zone model using borehole data. Through hydraulic testing and interpretation of borehole observations, DOE suggested that the volume and extent of the perched-water bodies at Yucca Mountain may vary greatly (SAR Section 2.3.2.2.2.4). The extent of perched water is inversely related to the extent of vitric CHn, which NRC staff reviews in the previous subsection on the southern portion of the repository. NRC considered the possibility that DOE underestimated the extent of perched water because this would lead to overestimates of average travel times, and thus potentially lead to underestimates of dose. NRC staff notes that it is unlikely that the extent of perched water is significantly underestimated because of geospatial constraints provided by the locations of boreholes and DOE's reasonable, but conservative, placement of the zeolitic contact with the vitric CHn.

In summary, DOE reasonably represented the influence of the CHn unit in the northern portion of the repository footprint because alternative models for the cause of perched water and the extent of perched water would either increase travel times or insignificantly reduce travel times.

Flow Patterns Below the CHn

DOE provided sparse observations related to hydrologic properties and flow patterns below the CHn in the northern and southern repository areas traversing the Prow Pass, Bullfrog, and CFu Tuffs. In its review of flow patterns below the CHn unit, NRC staff considered uncertainty caused by the sparseness of observations and data.

NRC staff reviewed the hydrologic characterization of layers below the CHn. To estimate the hydrologic properties of the lowermost layers to calibrate the site-scale unsaturated flow model, DOE supplemented the available information and observations with analog data from the PTn and TSw (SAR Section 2.3.2.3.5.3). NRC staff considered the effect on uncertainty on the basis of sparse data available from scattered boreholes that reach deep enough to cross these units. NRC staff notes that uncertainty in hydrologic properties and flow patterns in the Prow Pass, Bullfrog, and CFu Tuffs does not significantly affect calculations of dose, because (i) flow paths in the northern portion of the repository bypass these units, (ii) travel time in the southern portion of the repository is dominated by transport time in the Calico Hill nonwelded vitric tuff horizons, and (iii) assigned hydraulic properties lead to more focusing of flow and thus increase transport velocities through the highly sorbing matrix units in DOE's flow model.

NRC staff considered model support for the spatial distribution of zones of focused flow potentially provided by the pattern of water table temperatures. The NRC staff expects that large-scale zones of focused flow may depress the geothermal gradient in the unsaturated zone and perturb the temperature at the water table. Temperatures at the water table might reflect large-scale flow features, such as (i) localized and high flux rates predicted by the unsaturated zone model in faults; (ii) low flux reaching the water table below zeolitic rocks, which predominate in the northern half of the repository; or (iii) flux rates focused by decreasing areal extent of vitric CHn with depth, which predominates in the southern half of the repository. Alternative interpolations of water table temperature were presented in the SAR Figure 2.3.2-37, SNL Figure 6.3.1-7 (2008ag), and Sass, et al. (1988aa). Because the distribution of water table temperatures in any of the interpolations was not consistent with the large-scale spatial distribution of percolation in the DOE site-scale unsaturated zone flow model, DOE suggested that water table temperature was not a sensitive indicator to percolation rate, as outlined in DOE Enclosure 1 (2009cy). DOE suggested that multiple factors make it difficult to interpret potential relationships between temperature and percolation, including (i) uncertainty in ground surface temperature, (ii) thickness of the unsaturated zone, (iii) uncertainty in thermal conductivity of unsaturated zone units, (iv) influence of vertical groundwater flow in the saturated zone, and (v) uncertainty in the deep subsurface heat flux.

The locations of zones of focused flow are fixed in the DOE model. Because no other support was available to support predictions of the spatial distribution of flow reaching the water table, the NRC staff considered the consequence of uncertainty in the location of focused flow in the lower part of the unsaturated zone. DOE showed with sensitivity analyses that the exact locations of radionuclide release from the unsaturated zone to the saturated zone are not important in the performance assessment (SAR Section 2.3.2.3.5.4). Because the radionuclide release distribution is not important to performance, the distribution of focused flow also is not important to performance. On the basis of the DOE sensitivity analyses and the fact that less focusing would improve performance, the NRC staff notes that DOE reasonably accounted for the sparseness of data in model development for flow paths below the repository.

NRC staff reviewed water table position because of its effect on unsaturated zone transport length. Future, wetter climates affect flow below the repository in two ways: increased flow rates and water table rise. The increased percolation rates during future climates are evaluated as part of the site-scale unsaturated flow model in TER Section 2.2.1.3.6.3.2. The present-day water table is located in the dipping layers of the Prow Pass, Bullfrog, and Crater Flat Tuffs. Mineralogical and geochemical evidence suggests the water table occurred at higher elevations in the past. DOE found no evidence supporting higher perched water elevations underneath the repository; therefore, NRC only considered the uncertainty of a higher water table. DOE accounted for water table rise under future, wetter climate conditions by raising the location to a uniform 850-m [2,790-ft] elevation, which is approximately 120 m [394 ft] higher than the present-day estimate for the water table position. DOE stated (SAR Section 2.3.2.5.2) this rise in the water table is significantly greater than indicated by geologic evidence, which includes mineralogic alteration and isotopic ratios in secondary minerals, and flow modeling exercises with increased precipitation and recharge. Because a greater rise in the water table elevation reduces the transport path length for the unsaturated zone, and thus shortens travel times and reduces potential sorption capacity of the unsaturated zone, NRC notes that DOE acceptably incorporated the effects of future, wetter climates on flow paths below the repository.

2.2.1.3.6.3.6.3 Adequacy of Flow Fields for Transport

NRC staff reviewed the effect of uncertainty of flow fields on transport through the unsaturated zone. DOE passed the flow fields below the repository to the Unsaturated Zone Transport (SAR Section 2.3.8.5) portion of the DOE performance assessment. Overall, DOE considered advection to be the most important transport process in the unsaturated zone because the rate of water movement in the unsaturated zone largely controls radionuclide travel times, as outlined in SNL Section 6.1.2.1 (2007bj) and DOE Enclosure 6 (2009an). DOE also identified matrix diffusion and sorption as highly important for moderately to strongly sorbing radionuclides, particularly radionuclides with a short half-life that pass through a matrix unit, as described in DOE Enclosure 6 (2009an). DOE identified matrix diffusion and sorption as more important in the southern half of the repository because of the control from matrix transport in the CHn vitric facies, and more important for the 10,000-year period than the million-year period, as outlined in DOE Enclosure 6 (2009an). On the basis of these DOE assessments, the NRC staff focuses its review on the flow fields with respect to transport for nonsorbing and moderately to strongly sorbing dissolved radionuclides. NRC staff evaluates the repository performance with respect to unsaturated zone transport, including colloidal transport, in TER Section 2.2.1.3.7.3.2.4.

Flow path differences between the northern and southern portions of the repository influence the travel times of nonsorbing and sorbing radionuclides. DOE provided model results (SAR Figures 2.3.8-36 and 2.3.8-49) that exhibited the presence of three predominant types of transport pathways. NRC staff describes these as (i) fast transport for fracture releases occurs in the northern half of the repository, with mean travel times of years to centuries; (ii) moderately slow transport pathways for both matrix and fracture releases go through the southern half of the repository, with mean travel times of centuries to millennia; and (iii) slow transport through the matrix for radionuclides released into the matrix of the TSw tuff with mean travel times of millennia, with a small percentage transferring to the fracture system and reaching the water table more rapidly. The DOE ambient site-scale unsaturated zone model includes perching below the repository horizon in the northern half of the repository. In the DOE implementation, perching diverts fracture waters into faults and thereby creates a large difference in travel times for the northern and southern halves of the repository.

NRC staff first considered nonsorbing radionuclides. In DOE Enclosure 1 (2009am), several single-realization simulations with 30th and 50th percentile infiltration maps demonstrated transport properties for the unsaturated zone. Using DOE Enclosure 1, Table 4 (2009am), DOE calculated that total activity released from the unsaturated zone in the first 10,000 years after closure is 73 percent of total activity released from the Engineered Barrier System for Tc-99 (a nonsorbing radionuclide) in the northern half of the proposed repository and 78 percent in the southern half under the seismic ground motion scenario. DOE calculated that total Tc-99 activity released from the unsaturated zone in the first million years after closure is at least 98 percent of that released from the Engineered Barrier System for the igneous intrusion and seismic ground motion scenarios, regardless of release location.

The NRC staff notes that DOE reasonably represents flow in the unsaturated zone for nonsorbing radionuclides because the DOE transport model represents radionuclide transport processes through the unsaturated zone as not substantially reducing the activity of nonsorbing radionuclides released from the Engineered Barrier System.

The NRC staff next evaluates the extent to which the unsaturated zone flow fields affect DOE's performance assessment with respect to moderately to strongly sorbing radionuclides. DOE classified the porous matrix as either zeolitic, devitrified, or vitric and assigned all three classifications with sorptive capability. DOE explained that sorbing radionuclides are preferentially released to the fracture system as a result of sorption within the invert, as outlined in DOE Enclosure 1, Section 1 (2009am). As a result of the DOE release and flow models, which route fracture waters through the perched zone and into faults draining to the water table in the northern half of the repository, sorbing radionuclides predominantly bypass the matrix in the north. In the DOE flow model, both matrix and fracture waters pass into the matrix of the permeable and sorbing vitric CHn unit in the south.

The NRC staff notes that, for transport calculations of sorbing radionuclides in the northern portion of the proposed repository, DOE's conceptual model for perching represents a conservative approach for flow from the repository horizon to the water table, because almost all fracture waters and some matrix waters experience short travel times from the top of the perched zone to the water table. For the southern portion of the repository, DOE showed that reasonable changes to the hydraulic properties of the vitric CHn matrix will not significantly change the flow regime (SAR 2.3.8.5.2.2). The NRC staff notes that reasonable increases or decreases in hydraulic conductivity of the vitric CHn would minimally affect travel times of sorbing radionuclides. The NRC staff notes that, for transport calculations of sorbing radionuclides in the southern portion below the proposed repository, DOE's model represents a reasonable approach because the CHn vitric units have large matrix saturated hydraulic conductivity values so that matrix-dominated flows are likely.

2.2.1.3.6.3.6.4 Summary

On the basis of evaluations of the northern and southern portions of the proposed repository footprint, the NRC staff notes that the range of flow fields generated from DOE's site-scale flow model adequately represents model and data uncertainty for performance assessment calculations. In particular, the NRC staff notes that (i) the resulting flow fields are unlikely to overestimate radionuclide travel times from the proposed repository to the water table and (ii) different parameter value sets either would minimally affect travel times or would increase travel times. Because the flow fields are directly used as input in the transport model abstraction, NRC staff notes integration between the unsaturated flow and transport abstractions is reasonable. The NRC staff evaluates DOE's overall approach to transport modeling in TER Section 2.2.1.3.7.

2.2.1.3.6.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for unsaturated zone flow is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach in the areas discussed in the preceding section is reasonable for use in the Total System Performance Assessment.

2.2.1.3.6.5 References

Basagaoglu, H., S. Succi, C. Manepally, R. Fedors, and D. Wyrick. 2009aa. "Sensitivity of the Active Fracture Model Parameter to Fracture Network Orientation and Injection Scenarios." *Hydrogeology Journal*. Vol. 17. pp. 1,347–1,358.

Basagaoglu, H., K. Das, R. Fedors, R. Green, C. Manepally, S. Painter, O. Pensado, S. Stothoff, J. Winterle, and D. Wyrick. 2007aa. "Seepage Workshop Report." ML072980846, ML172980851, and ML072980850. San Antonio, Texas: CNWRA.

Beven, K. and A. Binley. 1992aa. "The Future of Distributed Models: Model Calibration and Uncertainty Prediction." *Hydrological Processes*. Vol. 6, No. 3. pp. 279–298.

BSC. 2004al. "Drift Degradation Analysis." ANL–EBS–MD–000027. Rev. 03. ACN 001, ACN 002, ACN 003, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004av. "*In-Situ* Field Testing of Processes." ANL–NBS–HS–000005. Rev. 03. ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004be. "Seepage Model for PA Including Drift Collapse." MDL–NBS–HS–000002. Rev. 3. ACN 001. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bf. "Thermal Conductivity of the Potential Repository Horizon." MDL–NBS–GS–000005. Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bt. "Mineralogic Model (MM3.0) Report." MDL–NBS–GS–000003. Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Burmeister, L.C. 1993aa. *Convective Heat Transfer.* 2nd Edition. New York City, New York: John Wiley & Sons, Inc.

Cussler, E.L. 1995aa. *Diffusion Mass Transfer in Fluid Systems*. 2nd Edition. New York City, New York: Cambridge University Press.

Das, K., S. Green, and C. Manepally. 2007aa. "FLOW–3D YMUZ2 Version 1.0 Users Manual." ML073030286. San Antonio, Texas: CNWRA.

DOE. 2010ai. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.5, Set 1 and (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (February 2) J.R. Williams to J.H. Sulima (NRC). ML100340034. Washington, DC: DOE, Office of Technical Management.

DOE. 2009am. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.7, Set 1." Letter (February 9) J.R. Williams to J.H. Sulima (NRC). ML090410352. Washington, DC: DOE, Office of Technical Management.

DOE. 2009an. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1." Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Washington, DC: DOE, Office of Technical Management.

DOE. 2009av. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 1. ML090700817. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bo. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (June 1) J.R. Williams to J.H. Sulima (NRC). ML091530403. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cb. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (June 5) J.R. Williams to J.H. Sulima (NRC). ML091590581. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cc. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (August 12) J.R. Williams to J.H. Sulima (NRC). ML092250006. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ct. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2, 2.3.3, and 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 2." Letter (July 20) J.R. Williams to J.H. Sulima (NRC). ML0920204130 and ML0920204141. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cx. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (December 14) J.R. Williams to J.H. Sulima (NRC). ML093490398. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cy. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2, 2.3.3, and 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 2." Letter (August 3) J.R. Williams to J.H. Sulima (NRC). ML092160370 and ML092160371. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Fedors, R., S. Green, D. Walters, G. Adams, D. Farrell, and S. Svedeman. 2004aa. "Temperature and Relative Humidity Along Heated Drifts With and Without Drift Degradation." ML042160472. San Antonio, Texas: CNWRA.

Flint, L.E. 1998aa. "Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain, Nevada." Water-Resources Investigation Report 97-4243. Denver, Colorado: U.S. Geological Survey.

Francis, N.D., Jr., S.W. Webb, M.T. Itamura, and D.L. James. 2003aa. "CFD Modeling of Natural Convection Heat Transfer and Fluid Flow in Yucca Mountain Project (YMP)." SAND 2002-4179. ACC:MOL20030906.0165. Albuquerque, New Mexico: Sandia National Laboratories.

Ghezzehei, T.A. 2004aa. "Constraints for Flow Regimes on Smooth Fracture Surfaces." doi:10.1029/2004WR003164. *Water Resources Research*. Vol. 40. p. W11503.

Green, S. and C. Manepally. 2006aa. "Software Validation Report for FLOW3D® Version 9.0." ML063050289. San Antonio, Texas: CNWRA.

Green, R.T., C. Manepally, R.W. Fedors, and M.M. Roberts. 2008aa. "Examination of Thermal Refluxing in *In-Situ* Heater Tests." ML083030097 San Antonio, Texas: CNWRA.

Incropera, F.P. and D.P. DeWitt. 2002aa. *Fundamentals of Heat and Mass Transfer*. 5th Edition. New York City, New York: John Wiley & Sons, Inc.

Incropera, F.P. and D.P. DeWitt. 1996aa. *Fundamentals of Heat and Mass Transfer*. 4th Edition. New York City, New York: John Wiley & Sons, Inc.
Kreith, F. and M.S. Bohn. 2001aa. *Principles of Heat Transfer*. 6th Edition. Pacific Grove, California: Brooks/Cole.

Kuehn, T.H. and R.J. Goldstein. 1978aa. "An Experimental Study of Natural Convection Heat Transfer in Concentric and Eccentric Horizontal Cylindrical Annuli." *Journal of Heat Transfer*. Vol. 100. pp. 635–640.

Kuehn, T.H. and R.J. Goldstein. 1976aa. "An Experimental Study and Theoretical Study of Natural Convection in the Annulus Between Horizontal Concentric Cylinders." *Journal of Fluid Mechanics*. Vol. 74, Part 4. pp. 695–719.

Leslie, B., C. Grossman, and J. Durham. 2007aa. "Total-system Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide." Rev. 1. ML072710060. San Antonio, Texas: CNWRA.

Liu, H.H., C. Doughty, and G.S. Bodvarsson. 1998aa. "An Active Fracture Model for Unsaturated Flow and Transport in Fractured Rocks." *Water Resources Research*. Vol. 34. pp. 2,633–2,646.

Manepally, C., S. Green, F. Viana, and R. Fedors. 2007ab. "Evaluation of In-Drift Heat Transfer Processes." ML071070508. San Antonio, Texas: CNWRA.

Manepally, C., A. Sun, R. Fedors, and D. Farrell. 2004aa. "Drift-Scale Thermohydrological Process Modeling—In-Drift Heat Transfer and Drift Degradation." CNWRA 2004-05. ML042160447. San Antonio, Texas: CNWRA.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

Or, D., M. Tuller, and R. Fedors. 2005aa. "Seepage Into Drifts and Tunnels in Unsaturated Fractured Rock." doi:10.1029/2004WR003689. *Water Resources Research*. Vol. 41. p. WR05022.

Painter, S., C. Manepally, and D.L. Hughson. 2001aa. "Evaluation of U.S. Department of Energy Thermohydrologic Data and Modeling Status Report." ML043080474. San Antonio, Texas: CNWRA.

Philip, J.R., J.H. Knight, and R.T. Waechter. 1989aa. "Unsaturated Seepage and Subterranean Holes: Conspectus and Exclusion Problem for Circular Cylindrical Cavities." *Water Resources Research*. Vol. 25, No. I. pp. 16–28.

Raithby, G.D. and K.G.T. Hollands. 1985aa. "Natural Convection." *Handbook of Heat Transfer Fundamentals*. 2nd Edition. W.M. Rohsenow, J.P. Hartnett, and E.N. Ganic, eds. New York City, New York: McGraw-Hill.
Richards, L.A. 1931aa. "Capillary Conduction of Liquids Through Porous Mediums." *Physics.* Vol. 1. pp. 318–333.

Salve, R. and T.J. Kneafsey. 2005aa. "Vapor-Phase Transport in the Near-Drift Environmental at Yucca Mountain." *Water Resources Research*. Vol. 41. p. W01012.

Sass, J.H., A.H. Lachenbruch, W.W. Dudley, Jr., S.S. Priest, and R.J. Munroe. 1988aa. "Temperature, Thermal Conductivity, and Heat Flow Near Yucca Mountain, Nevada: Some Tectonic and Hydrologic Implications." USGS Open-File Report 87-649. Denver, Colorado: U.S. Geological Survey.

SNL. 2008ab. "Features, Events, and Processes for the Total System Performance Assessment: Analyses." ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ai. "Postclosure Analysis of the Range of Design Thermal Loadings." ANL–NBS–HS–000057. Rev. 00. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008aj. "Multiscale Thermohydrologic Model." ANL–EBS–MD–000049. Rev. 03. ADD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008an. "Particle Tracking Model and Abstraction of Transport Processes." MDL–NBS–HS–000020. Rev. 02. ADD 02. Las Vegas Nevada: Sandia National Laboratories.

SNL. 2007aj. "EBS Radionuclide Transport Abstraction." ANL–WIS–PA–000001. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007az. "Simulation of Net Infiltration for Present-Day and Potential Future Climates." MDL–NBS–HS–000023. Rev. 01. AD 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bf. "UZ Flow Models and Submodels." MDL–NBS–HS–000006. Rev. 03. ACN 01, ERD 01, ERD 02, ERD 03. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bj. "Radionuclide Transport Models Under Ambient Conditions (U0060)." MDL–NBS–HS–000008. Rev. 02. CAN 002, AD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bk. "Abstraction of Drift Seepage." MDL–NBS–HS–000019. Rev. 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bl. "In-Drift Natural Convection and Condensation." MDL–EBS–MD–000001. Rev. 00. ADD 01. Las Vegas, Nevada: Sandia National Laboratories.

Stothoff, S.A. 2010aa. "Infiltration and Unsaturated Zone Confirmatory Analyses." Electronic Scientific Notebook 1005E. ML111110877. San Antonio, Texas: CNWRA.

Stothoff, S. and G. Walter. 2007aa. "Long-Term-Average Infiltration at Yucca Mountain, Nevada: Million-Year Estimates." CNWRA Report 09-003. ML072760607. San Antonio, Texas: CNWRA.

van Genuchten, M.Th. 1980aa. "A Closed-Form Equation for Predicting the Hydraulic Conductivity of Unsaturated Soils." *Soil Science Society of America Journal*. Vol. 44. pp. 892–898.

Zhou, Q., H.H. Liu, F.J. Molz, Y. Zhang, and G.S. Bodvarsson. 2007aa. “Field-Scale Effective Matrix Diffusion Coefficient for Fractured Rock: Results From Literature Survey.” *Journal of Contaminant Hydrology*. Vol. 93, Nos. 1–4. pp. 161–187.

(Intentionally Left Blank)

CHAPTER 10

2.2.1.3.7 Radionuclide Transport in the Unsaturated Zone

2.2.1.3.7.1 Introduction

This chapter provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) representation of radionuclide transport in the unsaturated zone. This is a key component in DOE's performance assessment for the proposed geologic repository at Yucca Mountain, Nevada, as identified in its Safety Analysis Report (SAR) Figure 2.3-2 (DOE, 2008ab). DOE's performance assessment analysis included the flow of water from precipitation falling on Yucca Mountain, its migration as groundwater through the unsaturated zone above and below the repository, and the flow of groundwater in the saturated zone to the accessible environment. Exposure to radionuclides in groundwater extracted by pumping is one of the principal pathways for radiological exposures to the reasonably maximally exposed individual and for releases of radionuclides into the accessible environment. Therefore, as required by 10 CFR 63.114, the performance assessment analysis included radionuclide transport in the unsaturated zone among those model components that significantly affect the timing and magnitude of transport for any radionuclides released from the repository. In SAR Section 2.3.8, DOE (i) described the features, events, and processes (FEPs) that DOE included to model the transport of radionuclides in groundwater in the unsaturated zone below the repository and (ii) provided the technical basis for DOE's implementation (or abstraction)[1] of the unsaturated zone transport model in the Total System Performance Assessment (TSPA) model. The NRC staff's evaluation focuses on the following processes, detailed in subsequent sections of this chapter, that DOE included in its SAR Section 2.3.8 as important for radionuclide transport in the unsaturated zone: (i) advection, because most of the radionuclide mass is carried through the unsaturated zone by water flowing downwards to the water table; (ii) sorption, because sorption in porous media in the southern half of the repository area has the largest overall effect on slowing radionuclide transport in the unsaturated zone; (iii) matrix diffusion in fractured rock, because matrix diffusion coupled with sorption slows radionuclide transport in the northern half of the repository area; (iv) colloid-associated transport, because radionuclides attached to colloids may travel relatively unimpeded through the unsaturated zone; and (v) radioactive decay and ingrowth, because these processes affect the quantities of radionuclides released from the unsaturated zone over time. The NRC staff's review of DOE's technical basis for excluding other FEPs is addressed in the Technical Evaluation Report (TER) Section 2.2.1.2.1 (Scenario Analysis).

DOE's radionuclide transport model abstraction for the unsaturated zone utilizes information on the magnitude and patterns of groundwater flow in the unsaturated zone and the flux of radionuclides released from the waste forms and engineered barrier systems (EBS). In turn, the unsaturated zone radionuclide transport abstraction provides information about the mass flux of radionuclides released to the saturated zone.

2.2.1.3.7.2 Evaluation Criteria

The NRC staff's assessment of model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for radionuclide transport in the

[1]The term "abstraction" is defined as a representation of the essential components of a process model into a suitable form for use in a Total System Performance Assessment. Model abstraction is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.

unsaturated zone, is guided by provisions in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff evaluation of inclusion or exclusion of FEPs is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose during the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The NRC staff review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Section 2.2.1.3.7, Radionuclide Transport in the Unsaturated Zone, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions, which provide guidance for the NRC staff's evaluation of DOE's abstraction of radionuclide transport in the unsaturated zone, are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

Consistent with a risk-informed approach, NRC staff review used the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of radionuclide transport in the unsaturated zone important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE and on NRC staff knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.7.3 Technical Evaluation

The NRC staff reviewed information in SAR Section 2.3.8 and references therein that described how DOE predicted the transport of radionuclides in the unsaturated zone below the repository. The NRC staff's technical evaluation focused on how DOE (i) represented transport-related geological, hydrological, and geochemical features of Yucca Mountain in the unsaturated zone transport abstraction, (ii) integrated the transport abstraction with other TSPA model components, and (iii) established the technical basis for modeling the major, risk-significant transport processes in DOE's process-level models and in the unsaturated zone radionuclide transport abstraction.

2.2.1.3.7.3.1 System Description and Model Framework

DOE used the Yucca Mountain site data in analyzing the downward flow of water in the unsaturated zone, through fractures, major faults, and rock matrix from the repository drifts to the water table. DOE used the same analytical framework for its modeling of unsaturated zone transport of radionuclides as for its site-scale unsaturated zone flow model (SAR Section 2.3.2): a three-dimensional representation of layered volcanic tuff units with specified geological and hydrological properties, in which water and radionuclides move through fractures in the rock, through the rock matrix, and between fractures and matrix. Major faults, which DOE assumed to provide fast transport pathways through the unsaturated zone, are represented in the model framework separately by a model with limited fracture–matrix interaction, as documented in SNL Section P21 (2008ag).

DOE simulated the transport of radionuclides as (i) dissolved species and as (ii) attached to mobile, colloid-sized particles. These two modes of transport are subject to various physical and chemical processes that affect radionuclide transport rates. DOE's conceptual model addresses how each of the transport-affecting processes influences the rate at which radionuclides travel through the unsaturated zone relative to the rate that water travels (SAR Section 2.3.8.2).

NRC Staff's Review

The NRC staff compared DOE's description of the unsaturated zone transport model framework in SAR Section 2.3.8 and references therein with the NRC staff's understanding of the Yucca Mountain natural system, obtained from extensive prelicensing field observations and independent analysis of the unsaturated zone transport processes, as identified in NRC Section 5.1.3.7 (2005aa) and Leslie, et al. (2007aa). The NRC staff notes that DOE's conceptual model appropriately includes FEPs that are expected to affect radionuclide transport in the unsaturated zone over the period of geologic stability (i.e., 1 million years). The NRC staff notes that DOE has provided a system description for radionuclide transport in the unsaturated zone because (i) DOE appropriately identified the Yucca Mountain site characteristics that produce vertical and lateral unsaturated zone groundwater flow pathways, lateral variability within layers (e.g., differences in the properties of the Calico Hills tuff in the southern part of the repository area compared to the northern area), and the presence of fault zones; (ii) DOE appropriately used Yucca Mountain site characterization data to develop geologic and hydrologic parameter values for specific rock units or to define ranges of values for these properties to address uncertainty about the natural variability of the system; and (iii) DOE adequately identified how and where the features of the unsaturated zone beneath the repository contributed to barrier capability in the performance assessment, as detailed in SAR Sections 2.3.8.1 and 2.1.2.3 and references therein and DOE Enclosure 1 (2009am). DOE adequately represented radionuclide transport in fracture-dominated and matrix-dominated unsaturated zone flow paths by using a model framework that, overall, is consistent with the conditions and assumptions of site-scale flow processes at Yucca Mountain.

2.2.1.3.7.3.1.1 Model Integration for the TSPA Code

DOE represented unsaturated zone transport as a model abstraction that simulates the transport of dissolved radionuclides and colloid-associated radionuclides through the unsaturated zone beneath the repository. This model generates breakthrough curves at the water table for the 27 aqueous species and 12 colloidal species that DOE determined were the most representative and risk significant (SAR Sections 2.3.8.6, 2.3.7.4.1.2, and 2.3.8.5.4). DOE simulated radionuclide transport in the abstraction with a residence-time particle-tracking technique in an external process model, FEHM, as identified in SNL Section 3.6 (2008ag). FEHM simulates flow and transport in three dimensions through fractured and porous rock. The three-dimensional volume through which the water and radionuclides travel is subdivided into a three-dimensional grid of cells, each of which is assigned fracture and matrix properties specific to each cell's spatial location. The particle-tracking technique determines the amount of time that a particle spends in each cell of the model and determines, on the basis of flow field information, which cell (fracture or matrix) the particle travels to next.

DOE integrated the unsaturated zone radionuclide transport abstraction with three other TSPA model components (SAR Figure 2.3.8-2): the EBS radionuclide transport abstraction (SAR Section 2.3.7.12), the site-scale unsaturated zone flow model (SAR Section 2.3.2.4.1), and the saturated zone radionuclide transport abstraction (SAR Section 2.3.9.3). Using flow conditions and concentration gradients at the boundary between the EBS and the unsaturated rock beneath the repository, the EBS transport abstraction calculates the radionuclide mass flux that enters the unsaturated zone fractures and rock matrix (SAR Section 2.3.7.12.3.2). Because the radionuclides travel more slowly in the rock matrix than in fractures (SAR Figure 2.3.8-49), DOE identified release of radionuclides from the EBS into the rock matrix as being a significant barrier mechanism (SAR Section 2.3.8.5.4). Accordingly, the NRC staff has evaluated DOE's technical basis for the flux-splitting submodel that calculates the release of radionuclides from the EBS into the rock matrix and into the fractures (TER Section 2.2.1.3.7.3.1.2).

DOE's site-scale unsaturated zone flow model passes flow field information to the unsaturated zone transport abstraction, such that radionuclide transport through the fractures and rock matrix in the model grid depends on the percolation (downward flow) fluxes provided by the flow fields. In particular, the flow fields generated by the site-scale unsaturated zone flow model provide the transport abstraction with spatial distributions of fracture-to-fracture, matrix-to-matrix, fracture-to-matrix, and matrix-to-fracture flow rates and moisture contents in the three-dimensional model framework, as detailed in SNL Section 6.3.9.2 (2008ag). During a calculated TSPA realization, the unsaturated zone transport abstraction receives flow field information from a sequence of up to four steady-state flow fields associated with different climate states (i.e., present-day, monsoon, glacial-transition, and post-10,000-year period) to account for future changes in percolation flux at specified points in time (SAR Section 2.3.8.5.3). In DOE's unsaturated zone transport abstraction model, the elevation of the water table beneath Yucca Mountain increases at the transition from the present-day climate state to a future, wetter climate state. The water table remains at the higher elevation for the remainder of the realization, effectively shortening the modeled thickness of the unsaturated zone transport path [SNL Section 6.4.8 (2008an); SNL Section 6.3.9.3 (2008ag)]. All other features of the unsaturated zone transport model grid and the sampled values of model parameters for the unsaturated zone transport abstraction remain constant throughout a TSPA realization.

The output of the unsaturated zone transport abstraction calculation provides time-dependent radionuclide mass flux as input to the saturated zone transport abstraction water table. The unsaturated zone transport abstraction groups the radionuclide mass fluxes into four collection regions and transfers the grouped mass fluxes to the saturated zone transport abstraction, which then initiates radionuclide transport in the saturated zone at an arbitrarily selected location for each of the four regions (SAR Section 2.3.8.5).

NRC Staff's Review

The NRC staff evaluated the information DOE provided in SAR Section 2.3.8 and in SNL Sections 6.3, 6.4, and 6.5 (2008an) and references therein about the development of the unsaturated zone transport abstraction and its integration with related model abstractions in TSPA calculations. The NRC staff verified, by examining results in SAR Section 2.4 and SNL Section 7.1[a] (2008ag), that the suite of radionuclides DOE used for the unsaturated zone transport calculations was consistent with the radionuclides that DOE's radionuclide screening analysis identified for transport in groundwater pathways, as provided in SNL Sections 6.2 and 6.3 (2007au).

The NRC staff compared DOE's integration of the unsaturated zone transport abstraction (SAR Section 2.3.8.5 and references therein) and the site-scale unsaturated zone flow model (SAR Section 2.3.2.4.1 and references therein). The NRC staff notes that DOE's representation of radionuclide transport in fracture-dominated and matrix-dominated unsaturated zone flow paths used well-documented and peer-reviewed FEHM modeling approaches (e.g., Doughty, 1999aa; Robinson, et al., 2003aa) that were consistent with the conditions and assumptions of the site-scale unsaturated zone flow model. In addition, DOE's integration of the unsaturated zone transport abstraction and the site-scale unsaturated zone flow model is reasonable because DOE's unsaturated zone transport abstraction used a model framework, technical bases, model properties, and assumptions that were consistent with the site-scale unsaturated zone flow model. The NRC staff's review of the site-scale unsaturated zone flow model is documented in TER Section 2.2.1.3.6.

The NRC staff evaluated DOE's integration of the unsaturated zone transport abstraction (SAR Section 2.3.8.5 and references therein) and the saturated zone transport abstraction (SAR Section 2.3.9.3 and references therein). The NRC staff notes that DOE's system description and model integration are reasonable for the unsaturated zone transport abstraction because DOE adequately described how its model assumptions about the transfer of radionuclides from the fractures and rock matrix of the unsaturated zone to fracture-dominated flow paths in the saturated zone were consistent with expected differences in site characteristics, flow model properties, and differences in scale between the unsaturated zone and the saturated zone transport paths.

2.2.1.3.7.3.1.2 Engineered Barrier System—Unsaturated Zone Boundary Condition

The release of radionuclides from the EBS is simulated in DOE's EBS transport abstraction and is used as input to the unsaturated zone transport abstraction. In DOE's model for the EBS, radionuclides are transported out of breached waste packages by advection (flow) or by diffusion, travel through the crushed tuff invert, and exit into the unsaturated rock at the base of the repository drift (SAR Section 2.3.7.12.3.2). At the model exit boundary, the EBS transport abstraction uses a submodel, which DOE termed the EBS–unsaturated zone interface submodel [SNL Section 6.5.2.6 (2007aj)], to distribute the radionuclides between the fractures and the rock matrix, according to modeled flow conditions and concentration gradients at the boundary. As stated in DOE Enclosure 6 (2009an), the overall result of the calculations in the EBS–unsaturated zone interface submodel is that most radionuclides released from waste packages in seeping drifts are transferred by advection into fractures, and most radionuclides released from nonseeping drifts are transferred by diffusion into the rock matrix.

In SAR Section 2.3.8.5.4 and DOE Enclosure 9 (2009am), DOE identified the initiation of transport in the low permeability rock matrix beneath the drifts as a mechanism that significantly delays the transport of radionuclides in the unsaturated zone. Accordingly, the NRC staff reviewed the technical basis for the submodel that DOE used to integrate the EBS transport abstraction and the unsaturated zone radionuclide transport abstraction in TSPA calculations. DOE provided information about the flux-splitting model in SAR Section 2.3.7.12.3.2; SNL Sections 6.5.2.5. and 6.5.2.6 (2007aj); SNL Sections 6.3.8 and 7.7.1[a] (2008ag); and DOE Enclosures 1, 9, 11, and 12 (2009am). The NRC staff has separately reviewed the technical basis and model properties for DOE's EBS transport abstraction and model properties in TER Section 2.2.1.3.4.3.5.

NRC Staff's Review

In the EBS–unsaturated zone interface submodel, DOE represents the near-field unsaturated zone as a localized, two-dimensional vertical array of overlapping fractures and rock matrix beneath the repository drift. The NRC staff compared DOE's model descriptions in SAR Sections 2.3.7.12.3.2 and 2.3.8.4.1 and verified that DOE's conceptual model for the EBS–unsaturated zone interface submodel was consistent with the modeling approach for fracture and matrix cells that DOE used for the unsaturated zone transport abstraction. The NRC staff also compared the conceptual basis of the EBS–unsaturated zone interface submodel with an independent model that relied on percolation flux and unsaturated matrix conductivity to simulate flux splitting, as identified in Leslie, et al. Chapter 11 (2007aa). The NRC staff confirms that DOE's approach resulted in releases to fractures and rock matrix that were similar to those for the alternate conceptual model. The NRC staff notes that DOE's flux-splitting model was adequately integrated with the unsaturated zone transport abstraction and the EBS transport abstraction because (i) DOE developed fracture and matrix water saturation and fluxes for the EBS–unsaturated zone interface submodel that were consistent with flow fields calculated by the unsaturated zone flow model, as identified in SNL Section 6.5.2.6 (2007aj); (ii) DOE chose values for rock properties and radionuclide transport parameters (e.g., sorption coefficients and effective matrix diffusion coefficients) in the submodel that were based on unsaturated zone transport processes and appropriate model properties for the Topopah Spring tuff subunits from DOE's unsaturated zone transport model; and (iii) DOE used transport properties for the crushed tuff invert for the EBS–unsaturated zone interface submodel that were based on an appropriate set of data from DOE's EBS transport model.

2.2.1.3.7.3.2 Unsaturated Zone Radionuclide Transport Processes

In DOE's unsaturated zone transport abstraction, the migration of radionuclides through the unsaturated zone is influenced by the transport-affecting processes of advection and dispersion, sorption, matrix diffusion, and colloid-associated radionuclide transport, as well as radioactive decay and ingrowth (SAR Section 2.3.8.1). Four of these processes—advection, dispersion, matrix diffusion, and colloidal transport—are transport mechanisms that move radionuclides from one location to another. In contrast, sorption may delay the transport of a radionuclide by attachment to stationary surfaces such as the rock matrix. Radioactive decay removes a radionuclide permanently from the system. Ingrowth is the replacement of a decayed radionuclide with a newly formed (daughter) nuclide, which may have different radioactivity and transport properties than the parent.

2.2.1.3.7.3.2.1 Advection and Dispersion

In DOE's unsaturated zone transport model and abstraction, advection refers to the transport of radionuclides, as either dissolved or colloid-associated phases, by the bulk movement of water. Overall, DOE considered advection to be the most important transport process in the unsaturated zone because, as DOE stated in SNL Section 6.1.2.1 (2007bj), the rate of water movement largely controls radionuclide travel times in the unsaturated zone. DOE coupled the advective transport of radionuclides with the bulk movement of water in fractures, in the rock matrix, and between fractures and matrix, using the groundwater flow rates and flow paths supplied by the site-scale unsaturated zone flow model, as detailed in SAR Section 2.3.8.5.2.1 and SNL Section 6.3.9.2 (2008ag). Because the unsaturated zone flow model predicts that water flows through the unsaturated zone at different rates in different rock units, the advective radionuclide transport rates vary correspondingly at different locations in the unsaturated zone. For example, in the fracture-dominated northern part of the repository area, DOE's unsaturated zone transport model predicts generally fast advective transport of radionuclides due to high modeled flow rates in fractures and faults. In the southern part of the repository area, advective transport of radionuclides in the unsaturated zone is slower due to low flow rates in the matrix-dominated flow system of the Calico Hills vitric tuff units (SAR Section 2.3.8.5.4).

The NRC staff reviewed information DOE provided about advection in SAR Section 2.3.8 and references therein. Because advective radionuclide transport in DOE's unsaturated zone transport abstraction depends directly on the flow field calculations supplied by DOE's unsaturated zone flow model abstraction, the NRC staff notes that the technical basis for the site-scale unsaturated zone flow model also provides the technical basis for DOE's representation of advective radionuclide transport in the unsaturated zone transport abstraction. On the basis of its technical evaluation of DOE's site-scale unsaturated zone flow model in TER Section 2.2.1.3.6, the NRC staff notes that DOE has provided a reasonable technical basis for modeling radionuclide transport by advection in the unsaturated zone transport abstraction.

DOE described dispersion as a spreading plume of dissolved radionuclides caused by localized differences in flow conditions, as identified in SAR Section 2.3.8.2.2.1 and SNL Section 6.3.9.1 (2008ag). As stated in SNL Section 4.1.6, AD01 (2008an), DOE did not identify dispersion as an important transport-affecting process at the scale of the unsaturated zone transport model. However, DOE chose to include a simple fixed-value longitudinal dispersion term in the transport model to support numerical analyses of breakthrough curves at the water table (SAR Section 2.3.8.5.2.2). To estimate a value for the dispersion term, DOE used results from saturated zone flow and transport tests at Yucca Mountain that were comparable in scale to site-scale unsaturated zone flow and transport paths (SAR Section 2.3.8.5.2.2).

The NRC staff reviewed the information DOE provided about dispersion in SAR Section 2.3.8 and references therein. From a risk-informed perspective, DOE provided an adequate technical basis for dispersion in the unsaturated zone transport abstraction because DOE provided adequate mathematical examples, field observations, and process-level modeling results to support DOE's statement that dispersion did not appreciably affect radionuclide travel times in the unsaturated zone transport calculations. In addition, DOE's representation of dispersion as a transport process was consistent with DOE's conceptual model of fracture and matrix flow conditions at Yucca Mountain; DOE estimated the value of the dispersion term appropriately from site-specific field tests that were representative of the expected scale of dispersion in the unsaturated zone transport abstraction; and DOE addressed data and model uncertainty for the dispersion term with simplifying assumptions that were appropriate for the minor effect of dispersion on radionuclide transport through the unsaturated zone.

2.2.1.3.7.3.2.2 Sorption

Sorption is a general term for chemical and physical processes that transfer a fraction of dissolved species to the surface of a solid phase. Depending on specific properties of the dissolved species, solid, and liquid, some dissolved species will sorb more readily onto solids than others will, and some will not sorb at all. DOE modeled sorption of dissolved radionuclide species in the unsaturated zone rock matrix but assumed that there was no sorption on fracture surfaces, except for those portions of the model framework that are designated as fault zones, as identified in SAR Section 2.3.8.5.2.3 and DOE Enclosure 2 (2009am). DOE also included sorption in modeling colloid-associated radionuclide transport, as discussed separately (TER Section 2.2.1.3.7.3.2.4).

In DOE's unsaturated zone transport model, sorption of radionuclides onto immobile rock surfaces slows the transport rate of radionuclides through the rock relative to the flow rate of water, a delaying effect that is called retardation (SAR Section 2.3.8.2.2.2). Sorption potentially can retard the transport of moderately or strongly sorbing radionuclides in the unsaturated zone for thousands of years or longer, contributing more significantly to unsaturated zone barrier capability than any other retardation process, as identified in DOE Enclosure 6 (2009an). In contrast, sorption of radionuclides onto mobile colloids, instead of onto immobile rock surfaces, may decrease the overall retardation effect.

DOE represented sorption in the unsaturated zone rock matrix with a sorption coefficient (K_d), an empirically determined or modeled value that represents the ratio of the sorbed-phase radionuclide concentration to the dissolved-phase radionuclide concentration. Low or zero values of K_d indicate that little or no sorption occurs; higher values indicate moderate or strong sorption, and therefore retardation. Factors that influence K_d values include the radionuclide chemistry and dissolved-phase concentration; the solution pH and major ion water chemistry; the temperature of the system; and the physical and chemical properties of the solid phase, including its surface area. Retardation by sorption is expressed in transport calculations by a retardation factor that depends on the value of the sorption coefficient and the physical properties (porosity and density) of the solid medium through which the radionuclide is transported. Retardation calculations assume that K_d does not vary with changes in radionuclide concentration, sorption and desorption reactions are fast relative to the flow rate, and bulk chemical composition of the water is constant (Davis and Curtis, 2003aa; Langmuir, 1997aa; Davis and Kent, 1990aa). In the unsaturated zone transport model and abstraction, DOE assumes that four radioelements are nonsorbing (carbon, chlorine, iodine, and technetium) and assigns a fixed value of K_d = 0 for each. For the remaining 11 radioelements modeled in unsaturated zone transport calculations (americium, cesium, neptunium, plutonium, protactinium, radium, selenium, strontium, thorium, tin, and uranium), DOE developed ranges and statistical distributions of K_d values for each radioelement and for each modeled rock unit from a combination of empirical data, process modeling, and professional judgment, as summarized in SAR Table 2.3.8-2. DOE detailed the K_d selection process in SNL Appendices A, B, I, and J and Addendum 1 (2007bj) and in DOE Enclosure 3 (2009am).

In terms of the barrier capability of the lower unsaturated zone (SAR Section 2.1.2.3), DOE attributed a higher overall importance to sorption than to any other transport process, as identified in DOE Enclosure 6 (2009an). Accordingly, the NRC staff has conducted a detailed review of the information DOE provided about sorption in SAR Section 2.3.8 and references therein, with a particular focus on (i) how DOE obtained data for the sorption model, (ii) how DOE addressed data and model uncertainty, and (iii) how DOE supported the sorption model as implemented in performance assessment calculations.

To obtain estimates of K_d values for sorption modeling, DOE grouped the various rock units below the repository into three rock types that have different sorption characteristics—zeolitized tuff, devitrified tuff, and vitric tuff (SAR Section 2.3.8.3.1). DOE measured sorption data from batch experiments that used site-specific crushed tuff samples and saturated zone water samples from two wells (J-13 and UE-25 p#1). DOE chose these water chemistries to bracket the major ion chemistry observed in pore waters and perched zone waters in the unsaturated zone, as provided in SAR Section 2.3.8.3.1 and SNL Section A4 (2007bj). DOE provided summaries of major ion chemistry (e.g., calcium, sodium, bicarbonate) for unsaturated zone pore waters, sampled by extraction from rock cores, and perched waters sampled by wells in locally saturated regions above the regional water table and compared the reported ranges with the two waters used in the sorption experiments, as identified in SNL Section A4 (2007bj). The NRC staff notes that for the purpose of estimating radionuclide sorption coefficients, DOE's use of the J-13 and UE-25 p#1 water chemistries adequately bounded the ranges reported for unsaturated zone water chemistries for major ions such as sodium, calcium, and bicarbonate at Yucca Mountain. For the long-lived actinides (americium, neptunium, plutonium, and uranium), DOE further characterized the effects of variability in geochemistry and mineral surface area using a non-electrostatic surface complexation modeling approach described in Davis, et al. (1998aa) and SNL Addendum 1 and Appendix A, Sections A7 and A8 (2007bj). This modeling approach is similar to independent models the NRC staff developed (e.g., Turner, et al., 2002aa) to examine the effects of broader chemistry ranges on several radionuclides.
In some cases, DOE supplemented the experimental and modeling sorption data with data from peer-reviewed scientific literature and reports prepared by other agencies {e.g., SNL Section A1[a] (2007bj)}. In the TSPA model, DOE sampled K_d values from the specified ranges to account for experimental uncertainty and variability in geologic conditions, including water chemistry and rock type, as detailed in SAR Table 2.3.8-2; SNL Appendices A, B, I, and J and Addendum 1 (2007bj) and DOE Enclosure 3 (2009am).

DOE identified mineral surface area and particle size as potential sources of data uncertainty related to the use of crushed tuff in experiments. The general DOE approach to addressing this uncertainty was to use batch experiments for a range of particle sizes and to bias the minimum and maximum limits for the K_d distributions toward lower (weaker sorption) values, as documented in DOE Enclosure 3, Table 1.1.2-1 (2009am). DOE referenced studies both from within and outside the DOE program in SNL Section 6.1.3.1 (2007bj). These studies indicated the effects of particle size on sorption are typically small except for the very fine (e.g., clay-sized) fraction. DOE's approach is reasonable because the approach properly identified potential sources of data uncertainty on the basis of site- and radionuclide-specific data and propagated the uncertainty through the unsaturated zone transport model abstraction. In addition, DOE's use of surface complexation modeling to extend the limited chemical conditions in the batch crushed tuff experiments supported DOE's technical basis for the upper and lower limits of sorption coefficients for the targeted actinides.

DOE addressed data uncertainty by obtaining empirical sorption data that assessed K_d variability as a function of time, radioelement concentration, atmospheric composition, water composition, particle size, and temperature. Although most of the data were gathered from batch sorption experiments, DOE also performed a limited number of confirmatory column tests on selected radionuclides that DOE had identified as important contributors to mean annual dose in previous performance assessment calculations. This is addressed in SAR Section 2.3.8.3.1 and SNL Table 4-1 (2007ba). In some cases, DOE indirectly addressed uncertainty related to radionuclide solubility by assigning low (conservative) K_d values. In selecting experimental data to inform the TSPA K_d distributions, DOE appropriately did not include data from experiments where the final radionuclide concentration may have exceeded a solubility limit. DOE outlined this approach in SAR Section 2.3.8.3.1, SNL Appendix A (2007bj), and SNL (2007ah).

In SNL Appendix A, Section A6 (2007bj), DOE reported that its modeled uncertainty distributions for K_d values tended to underpredict the effectiveness of sorption compared to the experimental distributions. In some cases, DOE reduced the upper bounds of the K_d distributions (specifically, those of cesium, plutonium, and radium) relative to the range indicated by available data to account for the possible effects of slow sorption kinetics for these elements, as identified in SNL Appendix A, Sections A8.4 and A8.6 (2007bj). By using low ranges of K_d values for sorption, DOE's transport model underpredicts retardation, resulting in faster radionuclide travel times through the unsaturated zone. With respect to the TSPA model abstraction of radionuclide transport through the unsaturated zone, this underprediction means that DOE takes less credit for sorption than the data indicated. The NRC staff notes that DOE adequately described how geochemical data were obtained, used, and interpreted to derive the K_d parameter distributions used to represent data uncertainty. Further, the NRC staff notes that where there is model uncertainty, DOE used assumptions and selected parameter values that would likely reduce the credit given to radionuclide sorption in its TSPA analysis.

In terms of model uncertainty and model support, the NRC staff notes that the empirical K_d modeling approach that DOE implemented is well established (e.g., Freeze and Cherry, 1979aa; Till and Meyer, 1983aa) and has been broadly used to describe radionuclide transport (e.g., Sheppard and Thibault, 1990aa). A potential model uncertainty associated with DOE's K_d approach is that individual K_d values are lumped parameters that do not explicitly take into account spatial and temporal variabilities or the role of specific surface-related processes that may affect radionuclide sorption. DOE addressed model uncertainty in its TSPA calculations by sampling K_d values stochastically from uncertainty distributions in which the distribution ranges were developed from expected system conditions. Rather than sample the K_d distribution independently for each radionuclide, DOE developed a correlation matrix for the 11 sorbing radioelements on the basis of their ranked sensitivities to six variables (pH, Eh, water chemistry, rock composition, rock surface area, and radionuclide concentration). DOE used this approach to approximate similarities in sorption behavior among radioelements and to ensure that transport behaviors were represented consistently within a single realization of the model, as detailed in SNL Appendix B, Section B1 (2007bj). In addressing model uncertainty, DOE did not take credit for sorption (i.e., $K_d = 0$) in fractures (fast flow paths), except for fault zones, and DOE implemented K_d uncertainty distributions for matrix sorption that in most cases predicted less sorption compared to measured distributions. The NRC staff notes that DOE's approach of taking no credit for sorption in fractures reduces the significance of model uncertainty of fractured systems on performance assessment. Excluding this process in fractures, which, if present, could contribute to waste isolation, results in radionuclide transport that is not underpredicted in fractures. Consequently, this approach is reasonable for addressing model uncertainty.

DOE developed information from natural analogs to provide qualitative comparisons for sorption model confidence building at the field scale (SAR Section 2.3.8.4.4). DOE did not apply the unsaturated zone transport abstraction to sorption modeling for these analog sites, and DOE did not use results from natural analog studies to inform the K_d distributions. Instead, DOE used general observations of sorption-related transport behavior to support the conceptual models (e.g., SAR Section 2.3.8.4.4.6). DOE also used observations from field sites at Busted Butte south of Yucca Mountain and alcove tracer tests with nonradioactive chemical homologues in the Exploratory Studies Facility to provide limited quantitative evaluations of sorption in the radionuclide transport model abstraction (SAR Section 2.3.8.3.3). The NRC staff notes that DOE's use of natural analogs to support model abstraction and uncertainty is reasonable for constraining sorption processes in unsaturated fractured rock. Furthermore, the NRC staff notes that it is reasonable to use site-specific sorption parameter values from Yucca Mountain to develop distributions instead of values from natural analogs.

In summary, DOE used consistent and appropriate assumptions to implement a linear sorption (K_d) model. DOE adequately described how it obtained, used, and interpreted experimental data with site-specific materials, alternative computer models, field tests, and natural analogs to provide a technical basis to support the TSPA model abstraction of radionuclide sorption. DOE considered appropriate radionuclides in the sorption model, using a linear K_d approach to sorption that is consistent with the other transport components of the TSPA model. DOE used site-relevant sorption data to address the anticipated effects of pH, Eh, major ion water chemistry, rock composition, rock surface area, and radionuclide concentration on radionuclide sorption concentration. DOE appropriately defined and documented the limitations of the K_d approach and used stochastically sampled K_d probability distributions and simplifying assumptions about the effectiveness of sorption to address model and data uncertainty. DOE considered appropriate geochemical and physical conditions in developing the K_d probability distributions, with either conservative or bounding assumptions that reduce the credit given to radionuclide sorption in the TSPA model. DOE appropriately considered alternative sorption modeling approaches and used them to support the technical basis for the K_d distributions. DOE adequately described the method used to assess the sensitivity of radioelement sorption behavior to variability in geochemical and physical conditions, and DOE appropriately used that method to correlate sorption characteristics among the radioelements, ensuring consistency among the sorption parameters for each model realization.

2.2.1.3.7.3.2.3 Matrix Diffusion

Diffusion is a physical process in which dissolved species or suspended particles move from a region of high concentration to a region of low concentration in accordance with the concentration gradient. DOE described matrix diffusion as a fracture–matrix interaction that uses diffusion to transfer radionuclides between fractures and the rock matrix. In DOE Enclosure 6 (2009an), DOE identified matrix diffusion as an important transport mechanism in the unsaturated zone transport abstraction, especially for strongly sorbing radionuclides, because it is the main process by which radionuclides can move from a fracture-dominated flow path into the matrix.

As DOE described in SNL Section 6.1.2.4 (2007bj), radionuclide transport by matrix diffusion in the unsaturated zone depends on (i) the matrix diffusion rate (i.e., the rate that a radionuclide can diffuse from water in a fracture into water in the pore spaces of the rock matrix) and (ii) the effective fracture–matrix interface, which is the area across which diffusion can occur. In turn, the matrix diffusion rate depends on (i) the radionuclide concentration gradient between fracture and matrix; (ii) the calculated water saturation of the rock; and (iii) the value of the effective matrix diffusion coefficient, which is a measure of how readily a particular radioelement diffuses through a tortuous pathway of interconnected pores in the rock matrix. To calculate the effective matrix diffusion coefficient, DOE multiplied the tortuosity coefficient by the free water diffusion coefficient. The tortuosity coefficient quantifies the reduction in diffusion rates resulting from the tortuous diffusion pathways through the rock. DOE determined this parameter from empirical data obtained from representative Yucca Mountain tuff samples. DOE developed standard normal cumulative probability distributions that were sampled stochastically in the TSPA analysis for each radioelement with respect to the individual model units (SAR Section 2.3.8.3.2; Reimus, et al., 2007aa).

DOE stated that not all connected fractures in unsaturated rocks actively conduct water (SAR Section 2.3.2.2.2.1), and, instead of uniform flow, individual fractures may have gravity-driven fingering flow that wets only a portion of a fracture surface (SAR Section 2.3.8.2.2.1). To adjust the size of the effective fracture–matrix interface area to account for the general observations of flow in the unsaturated fractures, DOE adopted the active fracture model (Liu, et al., 1998aa) for fracture–matrix interactions (SAR Section 2.3.2.2.2.1). In particular, DOE used an active fracture model parameter, *gamma*, and the modeled effective water saturation (i.e., the average water saturation of the connected fractures, adjusted by the unsaturated zone flow model for residual fracture saturation) to increase the modeled distance between flowing fractures. This reduced the size of the effective (i.e., wetted) fracture–matrix interface area for the unsaturated zone fracture–matrix interactions, thereby decreasing the amount of matrix diffusion and its capacity to retard radionuclide transport through the fractured rock.

In applying the active fracture model for flow field calculations in the site-scale unsaturated zone flow model (SAR Section 2.3.2.2.2.1), DOE used fixed values of the *gamma* parameter for individual model layers, estimated by flow model calibration, as detailed in SNL Section 6.3.2 (2007ad). In contrast, sensitivity analyses in SNL Section 6.6.4 (2008an) indicated that the radionuclide transport model calculations were more sensitive to *gamma* uncertainty values than the fluid flow model calculations. DOE reasoned that it would be inappropriate in the radionuclide transport calculations to assume that the *gamma* parameter was tightly constrained by the calibrated values from the fluid flow model. For radionuclide transport calculations, DOE instead sampled *gamma* values independently from an uncertainty distribution that was not limited to the calibrated fluid flow model values (SAR Section 2.3.8.5.2.4).

DOE's conceptual model of matrix diffusion in SNL Section C5 (2008an) assumes that matrix diffusion should be less effective in the unsaturated rocks than in the saturated rocks due to the reduced size of the wetted fracture–matrix interface area in unsaturated fractures. DOE cited field observations and numerical simulations of tracer migration in several large-scale transport experiments in the Exploratory Studies Facility to support its conceptual model of matrix diffusion in fractured, unsaturated rocks and use of the active fracture model for matrix diffusion calculations in the TSPA model (SAR Sections 2.3.8.3.3.2.1 and 2.3.8.3.3.3). DOE provided empirical observations and sensitivity analyses from field-scale experiments and process-level model analyses to address matrix diffusion model uncertainty. These were addressed in SNL Section 6.3.2 (2007ad); BSC Sections 7.4.1 and 7.4.2 (2004ag); BSC Section 6.12.2.4 (2004av); SNL Section 7.5 (2007bf); and Liu, et al. (2003aa).

In developing and supporting the matrix diffusion model, DOE acknowledged the impracticality of conducting large-scale transport tests to observe and measure the effects of fracture–matrix interactions in unsaturated rocks under natural conditions (SAR Section 2.3.8.3), and DOE cited uncertainties about the potential significance of scale-dependent transport processes in fractured rocks, as described in BSC Section 6.4.1 (2006aa) and Liu, et al. (2004aa). SNL Section 6.6.4 (2008an) identified that the size of the effective fracture–matrix interface area was the most uncertain term affecting radionuclide diffusion rates in the matrix diffusion model. To address model uncertainties about quantifying the effective fracture–matrix interface area for unsaturated zone transport calculations, DOE sampled the value of the active fracture model *gamma* parameter from a broad, uniform distribution that covered an intermediate range of 40 percent of all possible *gamma* values. The wide range of sampled *gamma* values produced a correspondingly wide range of results for radionuclide transport by matrix diffusion, as described in SNL Section 6.6.4 (2008an).

The NRC staff evaluated the information DOE provided on matrix diffusion in SAR Section 2.3.8, the hydrogeologic characteristics of the unsaturated zone at the Yucca Mountain site, and field and laboratory studies of fracture–matrix interactions in the unsaturated fractured rocks at Yucca Mountain and elsewhere, as detailed in NRC Section 5.1.3.7 (2005aa) and McMurry (2007aa). The NRC staff notes that DOE adequately described matrix diffusion and its integration into the unsaturated zone transport abstraction by coupling two transport-related physical phenomena—advection in fractures and diffusion in the rock matrix—in an approach that was consistent with DOE's dual-permeability model framework for unsaturated zone flow and radionuclide transport. DOE's conceptual model for the unsaturated zone matrix diffusion appropriately addresses differences between matrix diffusion in the unsaturated zone and the saturated zone by assuming that a comparatively smaller effective fracture–matrix interface area would be available for fracture–matrix interactions in unsaturated rocks.

In developing radionuclide-specific effective matrix diffusion coefficients for the unsaturated zone transport abstraction, DOE adopted a standard and widely used theoretical approach [Freeze and Cherry Section 3.4 (1979aa)] to estimate parameter values from laboratory measurements of diffusion properties in Yucca Mountain tuff samples. DOE addressed the uncertainty of the effects of fracture coatings on matrix diffusion by conducting diffusion experiments with paired core samples (i.e., samples with fracture coatings and without). In the tested sample pairs, mineral coatings on fracture surfaces did not impede diffusion rates (Reimus, et al., 2007aa). In addition, in BSC Section 5.2.1.1 (2004bi), DOE identified site characterization studies that showed secondary mineral coatings (e.g., calcite, oxides, clay minerals) are not abundant on fracture surfaces in Yucca Mountain tuffs, limiting their potential effects on matrix diffusion. Therefore, the NRC staff notes that DOE included sufficient data and addressed data uncertainty by measuring diffusion properties from an appropriate set of Yucca Mountain tuff samples, and DOE appropriately adapted the measured values to account for unsaturated conditions in the rock matrix.

The NRC staff compared DOE's descriptions of large-scale tracer test results with published results of other modeling studies and unsaturated zone field studies at Yucca Mountain and elsewhere. The comparison reflected DOE's statement that there is considerable uncertainty about the significance of fracture–matrix interactions in unsaturated rocks. Some transport studies identified fracture–matrix interactions as important (Zhou, et al., 2007aa; Liu, et al., 2004ab; Salve, et al., 2002aa; Hu, et al., 2001aa; Dahan, et al., 1999aa), and other studies did not (e.g., Pearcy, et al., 1995aa; Davidson, et al., 1998aa; Winterle and Murphy, 1999aa). Given the wide range of observations from various field studies, the NRC staff notes that DOE's use of a broad uncertainty distribution for the *gamma* parameter in the active fracture model is a reasonable treatment of model uncertainty for matrix diffusion in the unsaturated zone at Yucca Mountain. DOE conducted no unsaturated zone field experiments for model support specifically to evaluate DOE's matrix diffusion model under expected repository conditions, but DOE observed in several large-scale experiments in the Exploratory Studies Facility that tracer transport in fractured rocks took significantly longer than predicted by matrix diffusion in DOE's process-level models (SAR Sections 2.3.8.3.3.2.1, 2.3.8.3.3.2.2, and 2.3.8.3.3.3). DOE's numerical simulations of these tracer tests included numerical analyses based on the same model assumptions and the same broad range of *gamma* parameter values as DOE used for matrix diffusion in the unsaturated transport abstraction (SAR Section 2.3.8.4.4.4.3). This supports DOE statements in BSC Section 6.4.1 (2006aa) that despite model uncertainties, the TSPA transport calculations would not overestimate the potential of matrix diffusion to delay radionuclides in the unsaturated zone. DOE also provided the TSPA performance assessment results in SAR Section 2.4.2; SNL Section 7.7.1[a] (2008ag); and DOE Enclosure 6 (2009an) to demonstrate that the overall effect of matrix diffusion in delaying releases of radionuclides from the unsaturated zone was relatively unimportant, even for moderately and strongly sorbing radionuclides in the northern (fracture-dominated rapid transport) part of the repository area. Therefore, the NRC staff notes that from a risk-informed perspective, DOE's treatment of matrix diffusion model uncertainty and DOE's analyses to support the matrix diffusion model were reasonable because DOE's approach did not overestimate the effectiveness of matrix diffusion in retarding radionuclide transport.

In summary, DOE has provided an adequate description and technical basis for matrix diffusion in the unsaturated zone transport abstraction. DOE calculated effective matrix diffusion coefficients using parameters that DOE developed from an appropriate set of data from Yucca Mountain rock samples. DOE addressed data uncertainty in calculations of the effective matrix diffusion coefficients by including the natural variation of the rock properties in the range of parameter values. DOE adequately addressed model uncertainty about matrix diffusion in the unsaturated zone transport abstraction by sampling a broad, uniform distribution of values for the active fracture model *gamma* parameter. DOE addressed model uncertainty about the extent and importance of fracture–matrix interactions by varying the size and extent of the fracture–matrix interface area available for matrix diffusion over a large range of potential values, as detailed in SNL Section 6.6.4 (2008an), and by simulating the uncertain effect of uneven flow in space and time on transport rates in unsaturated fractures, as provided in SNL Section 6.3.9.2 (2008ag). DOE supported the assumptions and uncertainties of the modeling approach by demonstrating, through sensitivity analyses and by comparison with large-scale transport tests, that radionuclide transport in the unsaturated zone transport abstraction was less impeded by matrix diffusion than would otherwise be expected in a natural system.

2.2.1.3.7.3.2.4 Colloid-Associated Transport

Colloids are minute solid particles of any origin or composition that are suspended in a liquid. Colloids can form by many processes in natural or engineered systems—for example, by physical or chemical degradation of preexisting solid materials or by precipitation from a solution—or they can be of biological or geological origin (e.g., microbes, clay minerals). Colloids influence radionuclide transport in the unsaturated zone because the transport path and transport rate of radionuclides associated with a colloid (e.g., radionuclides attached by sorption to the colloid surface) are determined by the transport behavior of the colloid instead of by processes that might otherwise affect the transport rate of the radionuclide as a dissolved species (e.g., matrix diffusion, or sorption in the rock matrix). Compared to dissolved radionuclides, colloids migrate preferentially in fractures, where travel times tend to be fast, because the small size of matrix pore openings inhibits the transfer of colloidal particles from fractures to matrix.

DOE's conceptual model in SNL Section 6.3.9.1 (2008ag) defines two modes of colloid-associated radionuclide transport: reversible colloids, in which radionuclides are temporarily (reversibly) attached to colloids by sorption, and irreversible colloids, in which radionuclides are assumed to be permanently attached to or embedded in the colloid. According to SNL Section 6.5.3 (2007bi), the effectiveness of radionuclide transport by colloids depends on the transport characteristics of the colloids themselves, the concentration of colloids, and radionuclide sorption coefficients onto colloids and onto the immobile rock matrix, but the overall effect of colloid-associated transport of reversible colloids is to facilitate the transport of radionuclides through the system

DOE represented reversible colloid transport by modeling reversible sorption of dissolved radionuclides onto naturally occurring colloids in groundwater, using the same empirical K_d modeling approach that DOE used for reversible sorption in the rock matrix. DOE then applied an empirically determined colloid retardation factor, described in BSC Section 6.4.3 (2004bc), to account for colloid attachment and detachment processes in fractures that can hinder colloid movement in fractures. For simplicity, DOE assumed that all reversible colloids in the unsaturated zone are represented by the smectite clay mineral montmorillonite, a colloid-forming mineral in Yucca Mountain tuffs that has a high sorption capacity. DOE, in SNL Section 6.5.3 (2007bi), described how it modeled colloid-associated reversible sorption for six radioelements (americium, cesium, plutonium, protactinium, thorium, and tin) on the basis of their strong affinity for sorption onto montmorillonite under expected conditions in the unsaturated zone. DOE estimated the concentration of colloids in groundwater from data collected in saturated zone field studies from the Yucca Mountain area and from tabulated data for groundwater analyses elsewhere, as provided in SNL Table 6-21 (2008an). To address data uncertainty, DOE used the same estimated range of variability for groundwater colloid concentrations in the EBS, the unsaturated zone, and the saturated zone, but each transport abstraction sampled the range of values independently from the others in TSPA code simulations to account for the potential variability in groundwater colloid concentrations among the different environments, as identified in SNL Section 6.5.12 (2008an). DOE further addressed data uncertainty for reversible colloids by selecting ranges of montmorillonite sorption coefficients that emphasized large K_d values (i.e., strong sorption onto colloids), so as not to underestimate the effectiveness of radionuclide attachment to colloid surfaces, as described in DOE Enclosure 14 (2009am).

For irreversible colloids, DOE's colloid-associated transport model assumes that all irreversible colloids are generated within the EBS by the degradation of metals or wasteform materials, and the only radionuclides associated with irreversible colloids are isotopes of plutonium and americium (SAR Section 2.3.7.12.3.2). On the basis of field evidence for fast colloid transport in groundwater (e.g., Kersting, et al., 1999aa), DOE designated a small fraction (less than 0.2 percent) of the irreversible colloid flux as a "fast fraction" that is transported from the EBS to the accessible environment without any retardation. The rest of the irreversible colloid flux is subject to several potential retardation processes, including (i) fracture-related colloid attachment and detachment processes, as DOE detailed in SNL Section 6.5.13 (2008an); (ii) the direct release of irreversible colloids from the EBS into the low permeability rock matrix beneath the repository drifts, as described in DOE Enclosure 9 (2009am); and (iii) the advective transfer of irreversible colloids laterally from fracture flow paths into the rock matrix, subject to flow field conditions (i.e., matrix permeability large enough to accommodate the advective flux) and subject to colloid size exclusions at the fracture–matrix interface, as described in SAR Section 2.3.8.4.5.4 and SNL Sections 6.3.9.1 and 6.3.9.2 (2008ag). In SNL Section 6.3.9.1 (2008ag), and in SNL Section 6.5.9 (2008an), DOE also described a fourth retardation process, the matrix filtration (straining) of irreversible colloids at the interface between the matrix of one rock unit and the matrix of the underlying rock unit, resulting in the permanent immobilization of colloids in the unsaturated zone. DOE compared unsaturated zone breakthrough curves for irreversible colloids with and without matrix filtration in SNL, ERD 02, Section III (2008an) and observed that including matrix filtration as a retardation process diminished the flux of irreversible colloids out of the unsaturated zone by as much as 80 percent in the southern half of the repository area, as illustrated in SNL, ERD 02, Figure 6.6.2-6[c] (2008an). However, DOE's final TSPA model conservatively did not implement matrix filtration in the TSPA simulations, as explained in DOE Enclosure 11 (2009am).

The NRC staff reviewed DOE's technical basis for the colloid-associated transport model in the context of the NRC staff's independent understanding of colloid-associated transport modeling, colloid stability, and colloid transport properties in natural and engineered systems. As DOE noted in SAR Section 2.3.8.3, colloid transport mechanisms in unsaturated, fractured rocks are not well characterized by field studies. Accordingly, the NRC staff's review of DOE's technical basis for colloid-associated transport of radionuclides in the unsaturated zone focuses on how DOE addressed data and model uncertainty in developing parameter values and modeling colloid-associated transport processes. The NRC staff evaluated information DOE provided in SAR Section 2.3.8 and references therein, particularly SNL Section 7.7.1[a] (2008ag) and SNL (2008an). The NRC staff also considered additional information that DOE provided to clarify details of the colloid-associated transport model in DOE Enclosures 9 through 14 (2009am).

DOE provided a conceptual model for colloid-associated transport of radionuclides that appropriately incorporated observable phenomena to distinguish colloids from solutes, such as colloid sizes, colloid sorption properties, and colloid transport behavior in fracture-dominated flow systems. In addition, DOE's conceptual treatment of reversible and irreversible colloid-associated transport in the unsaturated zone transport abstraction was consistent with DOE's conceptual model for reversible and irreversible colloids in the EBS and saturated zone transport abstractions. Therefore, the NRC staff notes that DOE provided an adequate system description of the colloid-associated transport model and DOE integrated the model appropriately with other components of the unsaturated zone transport abstraction.

The NRC staff reviewed the data and methods that DOE used to estimate unsaturated zone transport parameters for colloids and notes that DOE compensated for a scarcity of unsaturated zone colloid transport data by using data from saturated zone groundwater analyses and Yucca Mountain saturated zone colloid transport tests to estimate unsaturated zone colloid properties. By incorporating the available site-specific data to set initial and boundary conditions for colloid properties, DOE's colloid-associated transport model adequately accounted for system variability and included sufficient data to describe colloids in the natural system. DOE addressed data uncertainty by (i) sampling large ranges for colloid-associated parameter values to account for data uncertainty about natural colloid properties and (ii) sampling the ranges of parameter values separately for the unsaturated zone and saturated zone transport abstractions to account for data uncertainty and spatial heterogeneity in the natural system.

DOE addressed model uncertainty for colloid-associated transport in the unsaturated zone by applying a number of simplifying assumptions about colloid-associated transport processes. The assumptions resulted in fast and relatively unimpeded transport of radionuclides associated with colloids compared to slower modeled travel times for the same radionuclides transported as solutes. In evaluating DOE's treatment of model uncertainty for reversible colloids, the NRC staff compared DOE's selection of ranges of sorption coefficients for montmorillonite colloids, detailed in SNL Table 6-22 (2008an), with DOE's selection of ranges of sorption coefficients for the unsaturated rock matrix, detailed in SNL Table 6-1[a] (2007bj). The NRC staff's comparison of values confirmed that for the radionuclides of interest, DOE's sorption coefficients for the montmorillonite colloids promoted stronger sorption onto colloids than onto the rock matrix. Therefore, the staff notes that including reversible colloid-associated transport in DOE's model does not overestimate radionuclide travel times in the unsaturated zone.

In evaluating DOE's treatment of model uncertainty for irreversible colloids, the NRC staff examined results of DOE unsaturated zone TSPA calculations for plutonium and americium radionuclides transported as dissolved species and as irreversible colloids, as illustrated by SAR Figure 2.4-108 and DOE Enclosure 10, Figures 3 and 4 (2009am). In addition, the NRC staff examined DOE sensitivity analyses in SNL, ERD 02, Section III (2008an), where DOE examined how irreversible colloid travel times through the unsaturated zone differed if the calculations included the effect of colloid filtration in porous rock matrix. From a risk-informed perspective, the NRC staff notes that DOE's treatment of model uncertainty is reasonable because DOE (i) used simplifying assumptions that resulted in faster transport of colloids than solutes in the unsaturated zone and (ii) took little or no credit in TSPA calculations for retardation processes such as the filtration of colloids at matrix–matrix interfaces that may significantly slow the transport of radionuclides associated with irreversible colloids compared to radionuclides transported as solutes.

In summary, DOE provided an adequate technical basis for the unsaturated zone colloid-associated transport model. DOE incorporated important processes and features of colloid transport that were consistent with the physical setting at Yucca Mountain. DOE adequately documented the conceptual and mathematical basis for the associated transport processes (e.g., retardation of colloids by attachment processes in fractures, reversible sorption of radionuclides onto colloids, colloid size exclusion processes at fracture–matrix interfaces, and unretarded colloidal transport), using an approach that was consistent with existing models for contaminant transport in fractured rocks in the literature (e.g., Sudicky and Frind, 1982aa). Because colloid transport properties in unsaturated, fractured rocks at Yucca Mountain and elsewhere were not well quantified by observations or experiments, DOE estimated unsaturated zone parameter values from site-specific saturated zone field and laboratory measurements. DOE appropriately addressed the large data uncertainty by sampling colloid parameter values probabilistically from large distribution ranges. With few empirical observations of unsaturated zone colloidal transport in field experiments or natural analogs to support the model abstraction output, DOE's modeling approach appropriately used a number of simplifying assumptions to compensate for uncertainty by taking little or no credit for colloid retardation processes in the unsaturated zone (SAR Sections 2.3.8.3.4 and 2.3.8.2.2.3).

2.2.1.3.7.3.2.5 Radionuclide Decay and Ingrowth

Radioactive decay is a general term for the processes by which unstable radionuclides spontaneously disintegrate to form a different nuclide that may or may not also be radioactive. DOE's particle tracking model in the unsaturated zone transport abstraction includes the loss of radionuclides over time due to radioactive decay and, where applicable, the model calculates the corresponding increase (ingrowth) of daughter radionuclides in decay chains, as described in SNL Section 6.4.4 (2008an). DOE assumed that upon radioactive decay of plutonium and americium in irreversible colloids, the decay chain daughters (e.g., uranium, neptunium) would be released from the irreversible colloid to migrate as dissolved species, with the exception of Pu-239 produced by radioactive decay of Am-243, which DOE assumed would remain irreversibly attached to the colloid (SAR Section 2.3.8.2.2.3).

The NRC staff examined DOE's radionuclide transport analyses in SAR Sections 2.3.8 and 2.4.2 and noted that DOE's results corresponded with expected changes in the transported inventory due to radioactive decay and ingrowth. The NRC staff notes that DOE's technical basis for including radioactive decay and ingrowth in the unsaturated zone transport abstraction is reasonable. The NRC staff did not conduct a detailed technical evaluation of DOE's model abstraction, because DOE calculated radionuclide decay and ingrowth using a standard mathematic decay equation, with radioactive decay constants that are known with precision (International Union of Pure and Applied Chemistry, 1997aa). DOE's representation of radioactive decay and ingrowth is reasonable because (i) DOE used a well-documented modeling approach with no significant uncertainties and (ii) DOE's model assumptions about the ingrowth-related transport behavior of decay chain radionuclides in irreversible colloids are consistent with DOE's model assumptions about the sorption behavior of the same radionuclides where they are associated with reversible colloids in DOE's model.

2.2.1.3.7.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for radionuclide transport in the unsaturated zone is consistent with the guidance in the YMRP. NRC staff also notes that the technical approach is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.7.5 References

BSC. 2006aa. "Analysis of Alcove 8/Niche 3 Flow and Transport Tests." ANL–NBS–HS–000056. Rev. 00. ACN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ag. "Conceptual Model and Numerical Approaches for Unsaturated Zone Flow and Transport." MDL–NBS–HS–000005. Rev. 01. ACN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004av. "*In-Situ* Field Testing of Processes." ANL–NBS–HS–000005. Rev. 03. ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bc. "Saturated Zone Colloid Transport." ANL–NBS–HS–000031. Rev. 02. ACN 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bi. "Yucca Mountain Site Description, Volume I: Sections 1–5." TDR–CRW–GS–000001. Rev. 02 ICN 01. ERD 01, ERD 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Dahan, O., R. Nativ, E.M. Adar, B. Berkowitz, and Z. Ronen. 1999aa. "Field Observation of Flow in a Fracture Intersecting Unsaturated Chalk." *Water Resources Research*. Vol. 35. pp. 3,315–3,326.

Davidson, G.R., R.L. Bassett, E.L. Hardin, and D.L. Thompson. 1998aa. "Geochemical Evidence of Preferential Flow of Water Through Fractures in Unsaturated Tuff, Apache Leap, Arizona." *Applied Geochemistry*. Vol. 13. pp. 185–195.

Davis, J.A. and G.P. Curtis. 2003aa. NUREG/CR–6820, "Application of Surface Complexation Modeling To Describe Uranium (VI) Adsorption and Retardation at the Uranium Mill Tailings Site at Naturita, Colorado." Washington, DC: NRC.

Davis, J.A. and D.B. Kent. 1990aa. "Surface Complexation Modeling in Aqueous Geochemistry. Mineral-Water Interface Geochemistry." M.F. Hochella, Jr. and A.F. White, eds. Mineralogy. Vol. 23. pp. 177–260.

Davis, J.A., J.A. Coston, D.B. Kent, and C.C. Fuller. 1998aa. "Application of the Surface Complexation Concept to Complex Mineral Assemblages." *Environmental Science & Technology*. Vol. 32. pp. 2,820–2,828.

DOE. 2009am. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.7, Set 1." Letter (February 9) J.R. Williams to J.H. Sulima (NRC). ML090410352. Enclosures (14). Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009an. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1." Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Enclosures (7). Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008ab. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 00. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Doughty, C. 1999aa. "Investigation of Conceptual and Numerical Approaches for Evaluating Moisture, Gas, Chemical, and Heat Transport in Fractured Unsaturated Rock." *Journal of Contaminant Hydrology*. Vol. 38, No. 1–3. pp. 69–106.

Freeze, R.A. and J.A. Cherry. 1979aa. *Groundwater.* Englewood Cliffs, New Jersey: Prentice-Hall, Inc.

Hu, Q., R. Salve, W.T. Stringfellow, and J.S.Y. Wang. 2001aa. "Field Tracer-Transport Tests in Unsaturated Fractured Tuff." *Journal of Contaminant Hydrology*. Vol. 51. pp. 1–12.

International Union of Pure and Applied Chemistry. 1997aa. *Compendium of Chemical Terminology*. 2nd Edition. A.D. McNaught and A. Wilkinson, eds. Oxford, United Kingdom: Blackwell Science.

Kersting, A.B., D.W. Efurd, D.L. Finnegan, D.J. Rokop, D.K. Smith, and J.L. Thompson. 1999aa. "Migration of Plutonium in Ground Water at the Nevada Test Site." *Nature*. Vol. 397, No. 6714. pp. 56–59.

Langmuir, D. 1997aa. *Aqueous Environmental Geochemistry*. Upper Saddle River, New Jersey: Prentice Hall.

Leslie, B., C. Grossman, and J. Durham. 2007aa. "Total-system Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide." Rev. 1. ML072710060. San Antonio, Texas: CNWRA.

Liu, H.H., G.S. Bodvarsson, and G. Zhang. 2004aa. "Scale Dependency of the Effective Matrix Diffusion Coefficient." *Vadose Zone Journal*. Vol. 3. pp. 312–315. Madison, Wisconsin: Soil Science Society of America.

Liu, H.H., R. Salve, J.S. Wang, G.S. Bodvarsson, and D. Hudson. 2004ab. "Field Investigation Into Unsaturated Flow and Transport in a Fault: Model Analyses." *Journal of Contaminant Hydrology*. Vol. 74. pp. 39–59. New York City, New York: Elsevier.

Liu, H.H., C.B. Haukwa, C.F. Ahlers, G.S. Bodvarsson, A.L. Flint, and W.B. Guertal. 2003aa. "Modeling Flow and Transport in Unsaturated Fractured Rock: An Evaluation of the Continuum Approach." *Journal of Contaminant Hydrology*. Vol. 62–63. pp. 173–178.

Liu, H.H., C. Doughty, and G.S. Bodvarsson. 1998aa. "An Active Fracture Model for Unsaturated Flow and Transport in Fractured Rocks." *Water Resources Research*. Vol. 34. pp. 2,633–2,646.

McMurry, J. 2007aa. "Overview of Field and Laboratory Studies of Unsaturated Zone Matrix Diffusion and Related Fracture–Matrix Interactions." ML072630430 San Antonio, Texas: CNWRA.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

Pearcy, E.C., J.D. Prikryl, and B.W. Leslie. 1995aa. "Uranium Transport Through Fractured Cilicic Tuff and Relative Retention in Areas With Distinct Fracture Characteristics." *Applied Geochemistry*. Vol. 10. pp. 685–704.

Reimus, P.W., T.J. Callahan, S.D. Ware, M.J. Haga, and D.A. Counce. 2007aa. "Matrix Diffusion Coefficients in Volcanic Rocks at the Nevada Test Site: Influence of Matrix Porosity, Matrix Permeability, and Fracture Coating Minerals." *Journal of Contaminant Hydrology*. Vol. 93. pp. 85–95.

Robinson, B.A., C. Li, and C.K. Ho. 2003aa. “Performance Assessment Model Development and Analysis of Radionuclide Transport in the Unsaturated Zone, Yucca Mountain, Nevada.” *Journal of Contaminant Hydrology*. Vol. 62–63. pp. 249–268.

Salve, R., J.S.Y. Wang, and C. Doughty. 2002aa. “Liquid-Release Tests in Unsaturated Fractured Welded Tuffs: I. Field Investigations.” *Journal of Hydrology*. Vol. 256. pp. 60–79.

Sheppard, M.I. and D.H. Thibault. 1990aa. “Default Soil Solid/Liquid Partition Coefficients, K_ds, Four Major Soil Types: A Compendium.” *Health Physics*. Vol. 59. pp. 471–482.

SNL. 2008ag. “Total System Performance Assessment Model/Analysis for the License Application.” MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008an. “Particle Tracking Model and Abstraction of Transport Processes.” MDL–NBS–HS–000020. Rev. 02. ADD 02. Las Vegas Nevada: Sandia National Laboratories.

SNL. 2007ad. “Calibrated Unsaturated Zone Properties.” ANL–NBS–HS–000058. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ah. “Dissolved Concentration Limits of Elements with Radioactive Isotopes.” ANL–WIS–MD–000010. Rev. 06. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aj. “EBS Radionuclide Transport Abstraction.” ANL–WIS–PA–000001. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007au. “Radionuclide Screening.” ANL–WIS–MD–000006. Rev. 02. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ba. “Site-Scale Saturated Zone Transport.” MDL–NBS–HS–000010. Rev. 03. ACN 01, AD 001. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bf. “Unsaturated Zone Flow Models and Submodels.” MDL–NBS–HS–000006. Rev. 03. ACN 01, ERD 01, ERD 02, ERD 03. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bi. "Waste Form and In-Drift Colloids-Associated Radionuclide Concentrations: Abstraction and Summary." MDL–EBS–PA–000004. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bj. “Radionuclide Transport Models Under Ambient Conditions.” (U0060). MDL–NBS–HS–000008. Rev. 02. CAN 002, AD 01. Las Vegas, Nevada: Sandia National Laboratories.

Sudicky, E.A. and E.O. Frind. 1982aa. “Contaminant Transport in Fractured Porous Media: Analytical Solutions for a System of Parallel Fractures.” *Water Resources Research*. Vol. 18. pp. 1,634–1,642.

Till, J.E. and H.R. Meyer. 1983aa. NUREG/CR–3332, “Radiological Assessment: A Textbook on Environmental Dose Analysis.” Washington, DC: NRC.

Turner, D.R., F.P. Bertetti, and R.T. Pabalan. 2002aa. "The Role of Radionuclide Sorption in High-Level Waste Performance Assessment: Approaches for the Abstraction of Detailed Models." *Geochemistry of Soil Radionuclides*. P.-C. Zhang and P.V. Brady, eds. Special Publication 59. Madison, Wisconsin: American Society of Agronomy. pp. 211–252.

Winterle, J.R. and W.H. Murphy. 1999aa. "Time Scales for Dissolution of Calcite Fracture Fillings and Implications for Saturated Zone Radionuclide Transport at Yucca Mountain, Nevada." Scientific Basis for Nuclear Waste Management, Boston, Massachusetts, November 30–December 4, 1998. Materials Research Society Symposium Proceedings 556. Warrendale, Pennsylvania: Materials Research Society. pp. 713–720.

Zhou, Q., H.H. Liu, F.J. Molz, Y. Zhang, and G.S. Bodvarsson. 2007aa. "Field-Scale Effective Matrix Diffusion Coefficient for Fractured Rock: Results From Literature Survey." *Journal of Contaminant Hydrology*. Vol. 93, Nos. 1–4. pp. 161–187.

CHAPTER 11

2.2.1.3.8 Flow Paths in the Saturated Zone

2.2.1.3.8.1 Introduction

This section of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the DOE representation of flow paths in the saturated zone within the context of DOE's performance assessment evaluation. The NRC staff reviewed information provided in the U.S. Department of Energy's (DOE) Safety Analysis Report (SAR) submitted on June 3, 2008 (DOE, 2008ab) and information provided in response to requests for additional information (RAIs).

Groundwater flow in the saturated zone is a key process in DOE's performance assessment evaluation for the proposed geologic repository at Yucca Mountain, Nevada. The performance assessment analysis summarized in the SAR includes the flow of water from precipitation falling on Yucca Mountain, its migration as groundwater through the unsaturated zone above and below the repository, and the flow of groundwater in the saturated zone through the controlled environment to the accessible environment. This groundwater is the principal means by which radionuclides released from the repository could be transported to the accessible environment. Exposure to extracted groundwater is one of the risk-significant pathways to the reasonably maximally exposed individual (RMEI); therefore, the performance assessment must include those components that significantly affect the timing and magnitude of transport for any radionuclides released from the repository.

DOE identified the saturated zone as a feature important to the capability of the lower natural barrier (SAR Section 2.1.1.3). Specifically, DOE indicated in Table 2.1-1 Expanded (DOE, 2009an) that the function of the saturated zone is to substantially reduce the rate of movement of radionuclides to the RMEI location by a combination of slow advective flow, long transport distance, and geochemical retardation of radionuclides. Hence, groundwater flow in the saturated zone is both the principal means for radionuclides to be transported to the location of the RMEI as well as an important function of the lower natural barrier to delay the movement of radionuclides along the path of travel to the location of the RMEI. Saturated zone groundwater flow, as described in SAR Section 2.3.9, includes the features, events, and processes (FEPs) that affect the movement of groundwater in the saturated zone to the accessible environment and their implementation (or abstraction)[1] in the Total System Performance Assessment (TSPA).

The saturated zone groundwater flow abstraction receives information about the magnitude and patterns of groundwater flow downward through the unsaturated zone. In turn, the saturated zone flow abstraction provides information about the direction, distance (flow paths), and amount (specific discharge) of groundwater flow to the saturated zone transport abstraction.

TER Section 2.2.1.1 provides the NRC staff's evaluation of DOE's identification and description of barriers and their capabilities as well as the consistency of these descriptions with the specific representations of these barriers in the TSPA and process-level models. The NRC staff's evaluation of those processes and characteristics most specific to radionuclide transport in the

[1]As used in the TER, the term "abstraction" refers to the representation of site characterization data, process-level models for FEPs, uncertainty and variability, and their overall integration in a (simplified) manner in the TSPA evaluation.

saturated zone is provided in TER Section 2.2.1.3.9. DOE's analysis of the effect of future climate change on water flow in the saturated zone is evaluated in this section while the NRC staff's evaluation of the nature of future climate change is presented in TER Section 2.2.1.3.5.

The other feature of the lower natural barrier DOE identified is the unsaturated zone below the proposed repository horizon (SAR Section 2.1.1.3). The NRC staff evaluates those features and processes related to the unsaturated zone below the proposed repository horizon in TER Sections 2.2.1.3.6 and 2.2.1.3.7.

2.2.1.3.8.2 Evaluation Criteria

The NRC staff's review of the model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for flow paths in the saturated zone, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4, 6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period [10 CFR 63.342]

The NRC staff review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Section 2.2.1.3.8, Flow Paths in the Saturated Zone, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of flow paths in the saturated zone are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff review used a risk-informed approach and the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of flow paths in the saturated zone important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination is based both on risk information provided by DOE and on NRC staff knowledge gained through experience and independent analyses.

2.2.1.3.8.3 Technical Evaluation

DOE analyzed the groundwater flow system in the vicinity of the proposed repository at Yucca Mountain to establish the direction and magnitude of water movement. DOE delineated the direction of water flow (flow paths) and computationally estimated the magnitude (specific discharge) of water flow using multiple groundwater flow models at different scales and degree of simplification. Specific discharge, in turn, is used to determine the timing of radionuclide transport.

The objective of the NRC staff's technical evaluation in this section is to determine the reasonableness of DOE's delineated flow paths and estimates of specific discharge (for both present and future conditions). The information evaluated in this section is from SAR Section 2.3.9 and from relevant supporting documents that are cited when referred to in this section of the TER.

TER Section 2.2.1.3.8.4 discusses the NRC staff's assessment of DOE's postclosure performance assessment for flow paths in the saturated zone.

2.2.1.3.8.3.1 System Description and Integration of Models Relevant to Flow Paths in the Saturated Zone

DOE used multiple models at different scales to describe and quantify portions of the saturated zone groundwater flow system in the vicinity of the Yucca Mountain site. The NRC staff evaluates the roles of these models and their interdependencies in this section. The site of the proposed repository at Yucca Mountain is within the Death Valley regional groundwater flow system located in the southern part of the Great Basin, which in turn constitutes a subprovince of the larger Basin and Range physiographic province. At the regional scale, the Death Valley groundwater system reflects the arid climatic conditions and the complex geology of Basin and Range flow systems (SAR Section 2.3.9.2.1).

Groundwater in the regional system generally flows from recharge areas at high altitudes to the regional hydrologic sink in the bottom of Death Valley (SAR Section 2.3.9.2.1). The regional groundwater flow pattern is also conceptualized as a series of shallow and localized flow paths superposed on deeper regional flow paths (SAR Section 2.3.9.2.1).

In the Yucca Mountain region, groundwater flows from generally north to south, following these regional flow patterns. A relatively small amount of recharge occurs in the immediate vicinity of Yucca Mountain migrating downward through the unsaturated zone to the saturated zone. Once in the saturated zone, groundwater flows through a volcanic rock aquifer in the northern portions of the general flow system, transitioning into an alluvial aquifer system in the southern portions of the Yucca Mountain region. Beneath both the volcanic and alluvial aquifers is a carbonate rock aquifer (SAR Section 2.3.9.2.1).

DOE indicated that at the regional scale, a significant amount of groundwater flows through the relatively permeable, laterally continuous, and thick carbonate aquifer, below volcanic and alluvial aquifers. On the basis of the regional geological framework and observations obtained from several drilled boreholes, DOE concluded that an upward hydraulic head gradient generally exists between the carbonate aquifer and the overlying volcanic and alluvial aquifers (SAR Section 2.3.9.2.2.4), which indicates that vertical groundwater movement, to the extent it occurs, is upward rather than downward.

DOE conceptualizes that the upward hydraulic gradient restricts groundwater flow paths originating from the proposed repository location to the shallower volcanic and alluvial aquifers, precluding radionuclides from entering the regional carbonate aquifer. DOE also believes the upward gradient will be sustained during future climates and water uses (SAR Section 2.3.9.2.2.4).

At the site scale, DOE stated that groundwater flow occurs from the recharge areas in the north, through the Tertiary volcanic aquifers into the valley-fill aquifer, and continues south toward the RMEI location. DOE used various site-scale saturated zone flow and transport models to predict groundwater flow paths and calculate the transport of radionuclides from their introduction at the water table below the proposed repository to the accessible environment. DOE summarized the interdependencies and information exchanges among these models (SAR Figure 2.3.9-1). The nominal case site-scale saturated zone flow model is conceptualized, and input parameters determined, on the basis of information derived from *in-situ* field tests, the U.S. Geological Survey Death Valley Regional Groundwater Flow System Model (DVRGFSM; which provides recharge and boundary conditions), DOE's site-scale hydrogeologic framework model, DOE's site-scale unsaturated zone flow model, and expert elicitation (SAR Section 2.3.9.1).

DOE's site-scale hydrogeologic framework model is a three-dimensional conceptual model of the spatial distribution of hydrogeologic units in the Yucca Mountain area. It covers an area of 1,350 km^2 [521 mi^2] and a thickness of about 6 km [3.7 mi] (SNL, 2007an). Direct input to the site-scale hydrogeologic framework model consists of hydrogeologic information from the DVRGFSM; DOE's site-scale geologic framework model, which was generated from DOE's investigations; and lithostratigraphic interpretations and coordinates from the Nye County Early Warning Drilling Program (NC–EWDP)[2] boreholes (SNL, 2007an). Within the site-scale hydrogeologic framework model, DOE divided the Yucca Mountain geologic units into five basic

[2]The Nye County Early Warning Drilling Program is a DOE-funded, Nye County-directed and -implemented hydrogeologic investigation program. DOE used information from this program to supplement its own investigations.

saturated zone hydrogeologic units on the basis of similar hydrogeologic properties: upper volcanic aquifer, upper volcanic confining unit, lower volcanic aquifer, lower volcanic confining unit, and lower carbonate aquifer (SNL, 2007an). DOE stated that certain characteristics affecting flow, primarily the porosity and permeability[3] of the hydrogeologic units, are highly variable. To represent discrete features and regions having distinct hydrological properties within the model domain, DOE identified and incorporated 10 hydrogeologic features into the flow model to represent such features as fault zones, hydrologic flow barriers, and zones of enhanced permeability (SNL, 2007ax).

DOE's site-scale saturated zone flow model is a three-dimensional finite-element numerical model that simulates groundwater flow in the area defined by the site-scale hydrogeologic framework model {i.e., 30 × 45 × 6 km [18.6 × 28.0 × 3.7 mi]}. DOE stated the flow model domain is sufficiently large to (i) assess groundwater flow and contaminant transport to the accessible environment, (ii) minimize boundary effects on flow magnitude and direction at Yucca Mountain, and (iii) include wells in the Amargosa Desert at the southern end of the modeled area (SAR Section 2.3.9.2.3.1). The site-scale saturated zone numerical flow model requires hydrogeologic information about the saturated zone flow system to predict flow magnitude and direction. DOE also provided this information, which includes groundwater flux from the system boundaries and physical attributes of the geologic media, in SAR Section 2.3.9.

DOE stated that the sources of surface recharge in the immediate vicinity of Yucca Mountain are precipitation and flood flows from Fortymile Wash and its tributaries (SAR Section 2.3.9.2.1). The site-scale saturated zone flow model obtains surface recharge information from DOE's site-scale unsaturated zone flow model over the area that lies directly below the site-scale unsaturated zone flow model domain. DOE used the 2004 version of the site-scale unsaturated zone flow model. However, DOE stated that an updated site-scale unsaturated zone flow model has been developed and is used in other parts of the TSPA (SAR Section 2.3.9.2.2.3). The NRC staff evaluates the impact of using an older version of the site-scale unsaturated zone flow model to estimate surface recharge over the area directly below the site-scale unsaturated zone flow model domain in TER Section 2.2.1.3.8.3.2.

DOE's three-dimensional site-scale saturated zone flow and transport abstraction model receives flow field information from the site-scale saturated zone flow model to generate 200 stochastic realizations of the flow field that reflect uncertainty in key parameters. DOE prepared the input for each flow realization by scaling all permeability values in the site-scale saturated zone flow model using a scaling factor sampled stochastically from the probability distribution of a groundwater-specific discharge multiplier. For permeability values within the volcanic aquifer hydrogeologic units, DOE also sampled stochastically the horizontal anisotropy ratio (the ratio in the permeability in one horizontal principal direction relative to the permeability in a different principal direction, usually vertical). The steady-state groundwater flow solution for each realization was established by running the site-scale saturated zone flow model (SAR Section 2.3.9.3.4.1). After completing the 200 realizations using the site-scale saturated zone flow model, the resulting 200 flow fields were input to the site-scale saturated zone flow and transport abstraction model. These flow fields provided the TSPA model with 200 radionuclide unit mass breakthrough curves at the RMEI location for 4 source subregions and 12 radionuclide groups, resulting in 9,600 breakthrough curves (SAR Figure 2.3.9-16). These breakthrough curves are evaluated in TER Section 2.2.1.3.9.

[3]Permeability is one of the most important characteristics of geologic media that affects the rate at which fluids can move through the medium.

The one-dimensional saturated zone transport abstraction model, which provides the transport simulation capability for radionuclide daughter products resulting from decay and ingrowth, uses a simplistic one-dimensional representation of the three-dimensional saturated zone flows. The one-dimensional saturated zone transport abstraction model consists of three pipe segments. The first pipe segment is 5 km [3.1 mi] long. The lengths of the second and third pipe segments are estimated from particle tracking results of the three-dimensional saturated zone flow and transport abstraction model. The variable lengths account for uncertainty in the location of the volcanic/alluvial aquifer contact. Average, homogeneous material properties and specific discharges are specified within each pipe. The average specific discharge along each pipe segment is calculated by dividing the flow path length by the 50th percentile of particle travel times in BSC Section 6.5.1 (2005ak).

The NRC staff evaluated DOE's description of the saturated zone groundwater flow system in the vicinity of the proposed repository and DOE's approach to integrate the multiple models used to quantify groundwater flow paths from the location of the proposed repository to the RMEI location. The NRC staff notes the description and approach is reasonable because

- DOE provided sufficient information to describe the aspects of hydrology, geology, physical phenomena, and couplings that are relevant to the regional and site-scale saturated zone groundwater flow system

- The roles of models and the interdependencies between different models were clearly identified and illustrated (e.g., DVRGFSM, site-scale hydrogeologic framework model, site-scale saturated zone flow model, site-scale saturated zone flow and transport model abstraction, and one-dimensional saturated zone transport model abstraction)

- Conditions and assumptions in models of saturated zone flow were consistently identified through the abstraction process (i.e., consistent with other interrelated model abstractions) and were consistent with information provided in the SAR

- Initial and boundary conditions used in the TSPA abstraction (i.e., site-scale saturated zone flow and transport model abstraction and one-dimensional saturated zone transport model abstraction) were consistent with the nominal case site-scale saturated zone flow model, which in turn was consistent with the DVRGFSM

The NRC staff's evaluation of the approach DOE used to calibrate the site-scale saturated zone flow model is documented in TER Section 2.2.1.3.8.3.2.

The NRC staff evaluated DOE's descriptions of relevant included FEPs and the manner in which they are included in the saturated zone flow models presented in SAR Section 2.3.9. The capability of the saturated zone to function as a barrier to delay radionuclide migration by slow advective flow and/or long transport distance depends on a number of processes and characteristics. DOE identified a number of general characteristics and processes important to the function of the saturated zone barrier including stratigraphy, water-conducting features, faults, fractures, properties of host rock and other (alluvial) units, groundwater flow in the geosphere (magnitude and direction of groundwater flow), advection and dispersion, climate change, matrix diffusion, and sorption (SAR Sections 2.1.1.3 and 2.3.9.1). Additionally, DOE stated these types of characteristics and processes of the saturated zone have been included in saturated zone flow and transport models presented in SAR Section 2.3.9 (Saturated Zone Flow and Transport).

2.2.1.3.8.3.2 Sufficiency of Baseline Data To Justify Models of Flow Paths in the Saturated Zone

DOE used site-specific data to develop and corroborate the conceptual model of groundwater flow in the saturated zone and to calibrate the site-scale saturated zone flow model. The site-specific data used include water-level measurements, *in-situ* hydrologic and tracer testing conducted in the vicinity of Yucca Mountain, regional hydrogeologic model predictions, and parameters from expert elicitation. The NRC staff evaluated the sufficiency of DOE's baseline data used to develop predictions of flow paths and groundwater flow rates in the saturated zone in this section.

The NRC staff evaluated the sufficiency of geological data that were relevant to the hydrogeologic framework used in the site-scale saturated flow model. DOE updated its site-scale hydrogeologic framework model to include stratigraphic information inferred from recently drilled Nye County wells. While DOE has added more hydrogeologic units to improve its hydrogeologic framework model, the contact geometry between volcanic tuff and valley-fill alluvial aquifers remains uncertain. Regarding repository performance assessment, the location at which groundwater flow moves from fractured volcanic rocks to alluvium is particularly significant because of the differences in the hydrologic properties between fractured volcanic units and the alluvium (SAR Section 2.3.9.2.1). In the performance assessment model, DOE stochastically varied the lateral extent of an alluvium uncertainty zone to propagate uncertainty associated with the tuff and alluvium contact. The NRC staff evaluates DOE's approach to treat uncertainty associated with the tuff/alluvium contact in TER Section 2.2.1.3.8.3.3.

The NRC staff reviewed the test methods and results of the hydraulic and tracer tests DOE conducted to corroborate its conceptualization of groundwater flow in the volcanic aquifers. These tests included several hydraulic and tracer tests (cross-hole tests) at the C-Wells Complex, consisting of boreholes UE-25 c#1, UE-25 c#2, and UE-25 c#3 (SAR Figure 2.3.9-7). DOE concluded that flow in the volcanic rock units mainly occurs through a well-connected fracture network and that large-scale horizontal anisotropy of aquifer permeability exists in the saturated zone, which is preferentially oriented in a north-northeast direction. The open-hole surveys done at the C-Wells Complex also yielded information on stratigraphy, lithology, matrix porosity, fracture density, and the major flowing intervals.

On the basis of the NRC staff review of DOE's hydraulic and tracer tests, the NRC staff notes DOE's hydraulic and tracer tests are appropriate to corroborate its conceptualization of groundwater flow in the volcanic aquifers because (i) the C-Wells Complex represents an appropriate location for inferring *in-situ* volcanic aquifer properties, (ii) DOE used appropriate techniques in conducting these tests, and (iii) the large-scale and cross-hole hydraulic tests yielded sufficient data to substantiate the applicability of the mathematical approach used with respect to representing site-scale groundwater flow in volcanic aquifers. DOE used the cross-hole well testing results to support its conclusions that (i) well-connected fracture networks exist in the volcanic aquifers and (ii) large-scale horizontal anisotropy of aquifer permeability exists in the volcanic aquifers, which is preferentially oriented in a north-northeast direction, consistent with the dominant fracture network observed at outcrops and in cores. The NRC staff notes that DOE's use of cross-hole testing is an appropriate approach to support DOE's assumptions about well-connected fracture networks and anisotropy of aquifer permeability because cross-hole tests tend to sample a larger number of possible flow paths and thus are appropriate for interpreting large-scale trends and effective permeability values. The results DOE obtained in these tests are consistent with its assumption of fracture-dominated flow in the volcanic aquifer system.

The NRC staff reviewed the results of the hydraulic and tracer tests DOE conducted to corroborate its conceptualization of groundwater flow in the alluvium. These tests included the hydraulic and tracer tests conducted at the Alluvial Testing Complex (centered at Nye County well NC–EWDP–19D), which is located along the simulated flow paths (SAR Section 2.3.9.2.4.2). DOE indicated that the saturated alluvium significantly reduces the movement of radionuclides to the accessible environment. The alluvial aquifer is generally conceptualized as a homogeneous hydrogeologic unit in the site-scale saturated zone flow model, except near the Fortymile Wash area. Testing results described in SNL Section 7.2.2.3 (2007ax) from the Alluvial Testing Complex and Nye County well 22S indicated that alluvium permeabilities vary over two orders of magnitude. DOE added a high-permeability zone—the Lower Fortymile Wash alluvial zone—to take into account "possible channelization" within the alluvium, as identified in SNL Section 6.4.3.7 (2007ba) and SAR Table 2.3.9-8. Similarly, DOE accounted for the effect of alluvium spatial heterogeneity on radionuclide transport by varying the effective porosity parameter in its performance assessment code. The NRC staff notes the data derived from alluvial hydraulic tests are reasonable for the intended use because (i) the Alluvial Testing Complex is on the simulated flow path and underlying alluvial structure is representative of the Lower Fortymile Wash alluvium and (ii) appropriate techniques were used to interpret hydraulic and tracer test data including, for example, type curve fitting of pump test data.

The NRC staff evaluated the sufficiency of water-level data DOE used to calibrate its site-scale saturated zone flow model. DOE used 161 time-averaged water-level measurements from 132 wells (multilevel measurements were obtained from some wells) within the model domain to (i) provide calibration targets for the site-scale saturated zone flow model, (ii) truncate the top of the flow model grid, and (iii) provide the boundary conditions around the perimeter of the model. DOE stated that water-level calibration targets represent steady-state values and reflect current water uses wherever pumping takes place (SNL, 2007ax). The NRC staff reviewed the coverage of water-level measurements that DOE collected and notes that the spatial and temporal coverage of water-level data is sufficient to calibrate the site-scale saturated zone flow model.

Water Flow Between the Lower and Upper Aquifers

DOE used water-level data from 17 wells to determine whether an upward gradient exists from the lower volcanic aquifer to the upper volcanic aquifer within the modeled domain (SAR Table 2.3.9-6). DOE concluded that (i) a notable upward vertical gradient appears to exist between the lower and upper volcanic aquifer at locations nearest Yucca Mountain and (ii) the direction of the vertical hydraulic gradient varies from location to location away from Yucca Mountain (SNL, 2007ax).

The NRC staff evaluated the methodology and the sufficiency of data DOE used to establish the boundary conditions of the site-scale saturated zone flow model. DOE derived constant-head boundary conditions from water-level data. In SNL Section 6.3.1.5 (2007ax), DOE stated that coverage of water-level measurements was insufficient to specify depth-dependent head boundaries. In SAR Section 2.3.9.2.3.1, DOE indicated that vertical gradients develop internally in the model domain in response to geohydrologic conditions and the calibrated model is capable of representing the upward vertical gradients observed between the deeper regional carbonate aquifer and overlying volcanic aquifers. DOE supplemented the SAR with a contour map of vertical hydraulic gradient distributions simulated by the site-scale saturated zone flow model, as shown in DOE Figure 1.1 (2009bc). The contour map indicated that the directions of the vertical hydraulic gradient developed internally by the site-scale saturated zone flow model are consistent with the observations in the wells penetrating the lower carbonate aquifer. On the basis of the additional information in the supplemental contour map DOE provided, the NRC staff notes that the site-scale flow model adequately represents the upward vertical gradient, both at the locations of the wells that penetrate the carbonate aquifer (UE-25 p#1 and Nye County well NC–EWDP–2DB) and in the rest of the model domain.

Recharge Data

The NRC staff reviewed the sufficiency of recharge data used in DOE's site-scale saturated zone flow model. DOE used recharge derived from a now-obsolete version of the site-scale unsaturated zone flow model. The NRC staff notes the impact of using an older version of the site-scale unsaturated zone flow model is small because the site-scale unsaturated zone component of recharge constitutes a small percentage (9 percent) of the total recharge within the domain of the site-scale saturated zone flow model. Surface recharge for other portions of the upper boundary within the domain of the site-scale saturated zone flow model was derived from the DVRGFSM model and measured stream losses along Fortymile Wash. The total surface recharge to the site-scale saturated zone flow model represents about 19 percent of the total inflow flux to the model domain. As a result, the recharge values from the unsaturated zone flow model are less than 2 percent of the total water budget for the model domain; therefore, the impact of uncertainty in the surface recharge from the site-scale unsaturated zone flow model to the site-scale saturated zone flow model is relatively small.

Vertical Anisotropy of Permeability

Vertical anisotropy of permeability is fixed at a ratio of 10:1 on the basis of information the expert elicitation panel provided (CRWMS M&O, 1998ac). In an alternative conceptual model, DOE considered the effect of vertical anisotropy on simulated flow paths. On the basis of previous experience with and knowledge of permeability anisotropy, the NRC staff notes that DOE's permeability vertical anisotropy ratio of 10:1 is in the generally accepted range and is consistent with a horizontally layered flow system like that in the Yucca Mountain region (Spitz and Moreno, 1996aa). In TER Section 2.2.1.3.8.3.4, the NRC staff evaluates the model uncertainty associated with the uncertainty in vertical and horizontal anisotropy ratios.

Site-Scale Model Calibration

DOE calibrated the site-scale saturated zone flow model using an industry-standard parameter estimation program (PEST) followed by manual adjustments. During calibration, DOE appropriately assigned higher weights to observation wells located on potential flow paths to the RMEI location. The NRC staff notes this is a reasonable approach. After calibrating the site-scale model using the parameter estimation program, manual adjustments were made to several zones to improve model match. The calibrated site-scale saturated zone flow model has a weighted root-mean-square (RMS) residual of 0.82 m [2.7 ft] (calculated using differences between observed and simulated heads). SAR Figure 2.3.9-13 shows locations of all water-level measurements and calibration residuals (i.e., differences between simulated and observed water levels at the calibration target locations). The magnitude of the weighted parameter estimation program calibration residual is reasonable for the scale of the model and the nature of the predictions made by the model (flow path direction and groundwater specific discharge) because the RMS residual value of 0.82 m [2.7 ft] is small compared to the total water-level elevation variation of more than 300 m [986 ft].

The NRC staff notes that the purpose of model calibration is to provide parameter estimates for a given conceptual model and is not intended to resolve uncertainties in model conceptualization (e.g., uncertainty in stratigraphy). Thus, DOE considered different alternative conceptual models to address model uncertainties. The NRC staff evaluates DOE's treatment of model uncertainty and alternative conceptual models in TER Sections 2.2.1.3.8.3.4 and 2.2.1.3.8.3.5.

The NRC staff has evaluated the data DOE used to develop and corroborate the conceptual model of groundwater flow in the saturated zone and to calibrate and validate the site-scale saturated zone flow model and notes the data are reasonable because

- DOE adequately summarized geological, hydrological, and geochemical data used to develop and implement models of saturated zone flow
- DOE used appropriate techniques correctly to conduct relevant well testing
- The description and justification of how the data were used, interpreted, and synthesized into model parameters were generally sufficient
- Sufficient data were collected to establish initial and boundary conditions
- Sufficient information was provided to substantiate that the site-scale saturated zone flow model is calibrated and applicable to site conditions

2.2.1.3.8.3.3 Uncertainty in Data Used in Models of Flow Paths in the Saturated Zone

Uncertainties in model input parameters may directly affect the advective flow rate of groundwater and lengths of groundwater flow paths predicted by DOE's nominal case site-scale saturated zone flow model. In the performance assessment evaluation, DOE incorporated the uncertainty in model parameter inputs by stochastically sampling values from probability distributions of the groundwater-specific discharge multiplier, horizontal anisotropy in permeability, flowing interval spacing and fracture porosity in the volcanic units, effective porosity in the alluvium, and longitudinal dispersivity, as identified in SNL Section 6.3 (2007ax). This TER section focuses on reviewing DOE's methodologies for developing probability distributions of the specific discharge multiplier and horizontal anisotropy in permeability. The NRC staff evaluates the other uncertain model parameter inputs relevant to radionuclide transport calculations in TER Section 2.2.1.3.9.

Specific Discharge Values and Multiplier

To incorporate uncertainty in specific discharge in model abstractions, DOE generated multiple realizations of the three-dimensional saturated zone flow field (refer to TER Section 2.2.1.3.8.3.1 for additional discussion on the interdependencies of the different saturated zone abstraction models). For each realization, DOE scaled (i) the values of recharge and all values of permeability simultaneously using a stochastically sampled specific discharge multiplier and (ii) the values of north-south and east-west permeability within the zone of volcanic rocks using a stochastically sampled horizontal anisotropy ratio.

DOE established a probability distribution for the groundwater-specific discharge multiplier, whose function is to capture the range in variability and uncertainty in the parameters that generate the specific discharge calculations. In turn the specific discharge calculations provide the basis from which groundwater travel times and radionuclide mass breakthrough curves are generated (SAR Section 2.3.9.2.3.3).

The uncertainty range and probability distribution for the groundwater-specific discharge multiplier was originally obtained through an expert elicitation process (CRWMS M&O, 1998ac). The expert elicitation panel suggested a truncated log-normal distribution ranging from 0.01 to 10. The median specific discharge derived from the expert elicitation process is 0.6 m/yr [2 ft/yr], as defined in Section 3.2 (CRWMS M&O, 1998ac). On the basis of recent tracer tests performed at the Alluvial Testing Complex and Nye County well cluster 22S, DOE reduced the range of uncertainty of the specific discharge multiplier using a Bayesian update procedure, where the range the expert elicitation panel supplied was assumed as a prior probability distribution and the estimated specific discharges from the Alluvial Testing Complex were used to estimate a log-normal likelihood function.

The NRC staff reviewed the Bayesian update procedure used by DOE. Because the Bayesian updating method requires that a dataset be composed of independent and identically distributed random samples, using this to reduce the specific discharge data set requires additional justification to support a meaningful Bayesian statistical analysis. In DOE Figure 1-1 (2009bc), DOE provided additional information explaining the rationale for using the Bayesian statistical procedure, stating that (i) each combination of interpretation method and effective porosity value provides an independent and equally likely outcome and (ii) the 12 data values follow approximately a log-normal distribution. After evaluating the additional information that DOE provided, the NRC staff notes that (i) DOE did not present a goodness-of-fitting statistical test to justify the log-normality of the 12 data, although the small sample size might preclude the meaningfulness of such a statistical test and (ii) DOE did not demonstrate the mutual independence of the estimation methods. Nevertheless, the NRC staff notes that uncertainty in specific discharge is appropriately bounded and propagated in DOE's performance assessment because (i) for each estimation method, DOE obtained three specific discharge estimates by assuming the underlying unknown porosity equal to the maximum, median, and minimum porosity values of a porosity distribution and, thus, likely bounded the estimation uncertainty related to the alluvium heterogeneity; (ii) the four estimation methods may have bounded the estimation uncertainty related to using each individual method alone; and most importantly (iii) realizations in DOE's performance assessment produce conservative transport times for nonsorbing solutes on the order of 10–100 years for the glacial-transition climate state, which the NRC staff notes does not result in an underestimation of the risk estimate (see also TER Section 2.2.1.3.8.3.4).

DOE used specific discharge estimates derived from alluvium testing to update the specific discharge multiplier that is subsequently applied to the entire flow model. DOE conceptualized fluid flow in volcanic tuff aquifers differently from the alluvium. The former is dominated by flow in well-connected fractures, while the latter is a porous medium. The Bayesian prior distribution the expert elicitation panel provided was based on tests performed at the C-Wells Complex, which were conducted in volcanic aquifers. However, variability of specific discharge in volcanic aquifers is likely smaller than that in the alluvium because of the difference in flow paths and permeability in the two types of aquifers. Thus, applying the same specific discharge multiplier distribution to both volcanic and alluvial aquifers does not result in underestimation of the overall uncertainty.

As a final result, DOE obtained a truncated log-normal distribution for the specific discharge multiplier that ranges from 1/8.93 to 8.93 (BSC, 2005ak). Because DOE chose to update the specific discharge multiplier instead of the specific discharge itself, the NRC staff questioned whether the mean specific discharge would change in light of data from Alluvial Testing Complex and Nye County well cluster 22S tracer testing. The staff also noticed that a shift in the estimated mean specific discharge is not explicitly propagated through Bayesian updating of the specific discharge multiplier. Thus, the NRC staff expressed concern that ignoring a possible change in the estimated mean specific discharge could lead to underestimation of the specific discharge uncertainty in the performance assessment abstraction models. In response to a request for additional information (RAI), DOE (2009bc) stated that (i) variations in mean specific discharge along the flow paths have been captured in the baseline, three-dimensional site-scale flow model and (ii) it is inappropriate to directly use specific discharge data from alluvial testing to update the expert elicitation estimates; instead, the update should be performed after normalization (i.e., the specific discharge multiplier). After evaluating this RAI response, the NRC staff notes that although the normalization process alone would not change the fact that specific discharge distribution in the alluvium is different from that in volcanic aquifers (also see the evaluation in the next paragraph) and although DOE calibrated the three-dimensional site-scale flow model against water levels but not specific discharges, DOE has demonstrated that simulated specific discharges are consistent with *in-situ* estimates (see TER Section 2.2.1.3.8.3.5). The NRC staff also notes that DOE's performance assessment model, using the updated specific discharge multiplier to address the glacial-transition climate state, produced conservative median transport times from the repository to the 18-km [11.2-mi] boundary for nonsorbing solutes on the order of 10–100 years (see TER Section 2.2.1.3.8.3.4). Therefore, the predicted radionuclide travel times are not inappropriately reduced, because of the approach DOE used to update the specific discharge multiplier.

Horizontal Anisotropy

The NRC staff reviewed the probability distribution that DOE established for the horizontal anisotropy of permeability. DOE stated in SNL Section 6.2.6 (2007aw) that hydraulic testing at the C-Wells Complex indicated significant flow anisotropy at larger scales in the fractured volcanic tuffs. In SAR Section 2.3.9.2.2.2.1, DOE indicated the horizontal anisotropy ratio is estimated using different methods and the ratio ranges from 3.3 to 17, with directionality orienting flow paths more north-south than east-west. The cumulative distribution function for the horizontal anisotropy ratio, which has lower and upper bounds of 0.05 and 20, respectively, is specified through a tabulated form in the performance assessment model.

The maximum anisotropy ratio (i.e., 20) is greater than the highest value the NRC staff independently estimated on the basis of site-specific data (Ferrill, et al., 1999aa). On the basis of the information DOE presented and the staff's independent estimate, the NRC staff notes the horizontal anisotropy ratio probability distribution is reasonable because it represents the level of uncertainty associated with the permeability anisotropy at the site scale.

Potentially Undetected Fast Flow Paths

The NRC staff reviewed DOE's conceptual model of the alluvial aquifers as homogeneous hydrogeologic units. The drilling Nye County conducted revealed significant spatial heterogeneity in the alluvium. In an independent analysis, the NRC staff conceptualized the Fortymile Wash alluvium as a gravel-dominated deposit, having lower permeability zones interstratified with higher permeability deposits (Sun, et al., 2008aa). The more permeable deposits, which have an estimated mean lateral length (i.e., the mean lengths of deposits or facies in the lateral directions) on the order of kilometers (1 km = 0.6 mi), might constitute fast flow pathways for radionuclide migration if well connected (Sun, et al., 2008aa). DOE stated that the suite of performance assessment transport simulations encompasses the range of behavior that would be obtained with a fault-based flow and transport model and other alternative conceptual models that explicitly model fast-flow paths (e.g., channeling in the alluvium) (SAR Section 2.3.9.2.3.5). DOE stated that the potential impacts of undetected features on groundwater flow are incorporated in the site-scale saturated zone flow and transport abstraction model through parameter distributions, which ultimately propagate to the TSPA model through variation in radionuclide breakthrough curves (BSC, 2005ak). The key parameters DOE used to assess the effect of undetected features on radionuclide transport are (i) specific discharge, (ii) porosity, (iii) flowing interval spacing in the volcanic rocks, (iv) longitudinal dispersion, (v) horizontal anisotropy of permeability, (vi) alluvial bulk density, and (vii) sorption coefficients for the nine classes of radionuclides modeled in both the alluvium and volcanic units (BSC, 2005ak). On the basis of the NRC staff review of (i) DOE's characterization and representation of parameter uncertainty in specific discharge and effective porosity through corresponding probability distribution functions and (ii) the travel times DOE's performance assessment code predicted, the NRC staff notes DOE's sampling from probability distributions of specific discharge multiplier and effective porosity addressed the worst-case scenario resulting from potentially undetected fast-flow paths.

Volcanic and Alluvial Aquifer Contact Zone

DOE introduces an alluvium uncertainty zone in the TSPA model to treat uncertainty associated with the contact location between volcanic aquifer and the alluvium. On the basis of drilling records, DOE conceptualizes the uncertainty zone as a quadrilateral area in which the boundary between volcanic units and alluvium is randomly varied among realizations. The boundaries of the alluvium uncertainty zone are determined for a particular realization by the parameters FPLAW (western boundary) and FPLAN (northern boundary) in DOE's one-dimensional saturated zone transport abstraction model. These parameters have uniform distributions from 0.0 to 1.0, where a value of 0.0 corresponds to the minimum extent of the uncertainty zone and 1.0 corresponds to the maximum extent of the uncertainty zone in a westerly direction and northerly direction, respectively (SNL, 2007ax). Thus, in the one-dimensional transport abstraction model, the flow path length of each pipe segment varies as a function of the horizontal anisotropy, the western boundary of the alluvial uncertainty zone, and the region from which the radionuclide source originates beneath the repository. The NRC staff has previously reviewed well drilling records from various phases of the NC-EWDP (Winterle and Farrell, 2002aa; Sun, et al., 2008aa). On the basis of these reviews, the NRC staff notes that (i) DOE reasonably bounded the extents of the alluvium uncertainty zone and (ii) the uniform distributions defined for FPLAW and FPLAN reasonably propagate uncertainties associated with the actual geometry of the volcanic and alluvium contact.

DOE used an expert elicitation process to obtain a probability distribution for the specific discharge multiplier (now considered prior distribution) and vertical anisotropy ratio. The NRC staff reviewed the information the expert elicitation panel provided. The NRC staff notes that the panel's probability distributions for specific discharge multiplier and vertical anisotropy ratio are reasonable. An overall NRC evaluation of DOE's expert elicitation procedures is provided in TER Section 2.5.4.

The NRC staff evaluates DOE's consideration of model uncertainty and model support in TER Sections 2.2.1.3.8.3.4 and 2.2.1.3.8.3.5.

The NRC staff notes that the data uncertainty characterization and representation in DOE's site-scale saturated zone flow and abstraction models are reasonable for the following reasons:

- DOE adopted parameter values, assumed ranges, probability distributions, and bounding assumptions that reasonably account for uncertainties and variabilities.

- DOE incorporated the hydrologic effect (e.g., water table rise) of potential climate change, on the basis of a reasonably complete search of paleoclimate data, using scaling factors for different climate states.

- The uncertainties associated with included features and events pertaining to groundwater flow in the saturated zone were represented in the model abstractions.

- Results from the expert elicitation panel defining the specific discharge multiplier and vertical anisotropy ratio probability distributions are reasonable.

2.2.1.3.8.3.4 Uncertainty in Flow Paths in the Saturated Zone Models

In this section, the NRC staff evaluates the alternative conceptual models DOE used to assess model uncertainties for the saturated zone flow paths, as presented in SAR Section 2.3.9.2.3.4. DOE used five alternative conceptual models to assess the significance of model uncertainties of certain features and processes in the abstraction, including (i) vertical anisotropy, (ii) horizontal anisotropy, (iii) permeability in the northern high-gradient region of Yucca Mountain, (iv) increased vertical permeability of the Solitario Canyon Fault, and (v) climate-induced water table rise.

The site-scale saturated zone flow model that DOE used to develop the performance assessment abstraction uses a 10:1 anisotropy ratio for horizontal-to-vertical permeability in volcanic and valley-fill alluvium units. This vertical anisotropy ratio was originally suggested from an expert elicitation panel (CRWMS M&O, 1998ac) because reduced vertical permeability is inherent in any layered groundwater flow system. To test whether this assumption leads to any systematic bias, DOE considered an alternative model with vertical permeability equal to the horizontal permeability. This alternative model resulted in a 28 percent increase in calculated specific discharge at a location 5 km [3.1 mi] downgradient from the proposed repository boundary and also resulted in a near doubling of the weighted RMS calibration error. Because this alternative model results in a degraded calibration and the inclusion of vertical anisotropy is considered more representative of the layered system, the model with the 10:1 vertical anisotropy ratio is the only one DOE used to develop the model abstraction. The NRC staff notes that DOE's analysis reasonably demonstrated that it is not necessary to consider alternative models with different vertical anisotropy ratios, because the potential effect on specific discharge is not significant compared to the range of uncertainty already considered in DOE's performance assessment model and the deleterious effects on model calibration statistics.

The NRC staff reviewed the alternative model DOE used to demonstrate the sensitivity of the nominal case site-scale saturated zone model to horizontal anisotropy. DOE's analysis demonstrated that removal of horizontal anisotropy (i.e., assuming isotropic horizontal permeability) results in a 31 percent decrease in modeled specific discharge rates across the 5-km [3.1-mi] boundary and shifts the flow paths eastward. The effect on model calibration error is negligible. This analysis demonstrated that both isotropic and anisotropic cases are relatively consistent with the observations in calibration wells. On the basis of this analysis, DOE included a range of horizontal anisotropy ratios for saturated zone flow and transport model abstraction. As discussed in TER Section 2.2.1.3.8.3.3, the NRC staff notes that the range of parameter uncertainty considered for horizontal anisotropy of permeability is appropriate.

The NRC staff reviewed another alternative modeling analysis by DOE in which DOE removes the large hydraulic gradient north of the proposed repository area by increasing permeability in this region. This alternative model results in a 15-fold increase in calculated specific discharge 5 km [3.1 mi] downgradient from the proposed repository and an eightfold increase in RMS calibration error. On the basis of this result, DOE concluded that, although the cause of the high gradient is not entirely certain, it is nevertheless important to represent this feature in the model. Similar to DOE's conclusion, the NRC staff considers that any modeling analysis that does not include low-permeability structural features to reproduce the large hydraulic gradients DOE measured north and west of the repository area would not be consistent with those measured gradients. DOE, therefore, appropriately included these features in the site-scale saturated zone flow model used to develop its performance assessment abstraction.

The NRC staff evaluated DOE's alternative model used to examine the potential effects of vertical anisotropy in the permeability of the Solitario Canyon fault. Such anisotropy could support along-fault flow or upward flow at the fault while acting as a barrier to cross-fault flow. The result of the analysis shows an insignificant effect on simulated water levels or flow paths; therefore, the NRC staff notes that incorporating this alternative model in the abstraction is not necessary, because it would not significantly affect the performance assessment results.

The site-scale saturated zone flow model DOE used to develop the abstracted flow paths for the performance assessment does not consider explicitly the effect of an elevated water table under future, wetter climate conditions. Instead, DOE incorporates the effect of long-term climate change by applying a scaling factor to instantly increase the volumetric flow rate or specific discharge. The scaling factors for monsoonal and glacial-transition climatic conditions are 1.9 and 3.9, respectively. To demonstrate that this simplified approach for including effect of climate change does not bias results, DOE provided an alternative evaluation of the potential effects of water table rise on abstracted flow paths. On the basis of an estimated increase in specific discharge by a factor of 3.9 for the glacial-transition climate state, DOE estimated the increased hydraulic gradient necessary to drive this increased groundwater flow would result in a water table rise of approximately 20 m [66 ft] at the southern end of the model area that gradually increases to about 50 m [160 ft] in the area below the proposed repository location and as much as 100 m [328 ft] in areas north of the repository. Projecting this linear increase in water table elevation onto the hydrogeologic framework model indicates that elevated flow paths could travel a greater proportion of distance through the lower permeability Calico Hills formation in the volcanic tuffs, which could result in longer travel times. DOE, therefore, concluded that using present-day water table elevations combined with a scaling factor approach to increase specific discharge estimates for future climates sufficiently approximates the performance-affecting aspects of future, wetter climate conditions. On the basis of the staff's examination of DOE's alternative evaluation of the potential effects of water table rise on abstracted flow paths, the NRC staff notes DOE's treatment of the effects of future climate on saturated zone flow is reasonable. Any bias introduced by excluding climate-induced water table rise would only lead to slower groundwater flow and would not significantly alter groundwater flow paths. The staff also notes the conclusion of DOE's evaluation is consistent with an independent Center for Nuclear Waste Regulatory Analyses analysis (Winterle, 2005aa), which indicates an elevated water table would not significantly affect flow paths from beneath the proposed repository area.

In SAR Section 2.3.9.2.3.5, DOE provided qualitative consideration of several additional model uncertainties that could affect estimates of specific discharge. These considerations and the NRC staff's review are summarized in the following paragraphs.

The NRC staff evaluated DOE's treatment of uncertainty in the hydrogeologic contact surfaces (as in the hydrogeologic framework model) represented in the model. DOE explained that horizontal contact-surface uncertainty would have a lesser effect on specific discharge compared to uncertainty of contact surfaces in the vertical direction. DOE concluded that the potential effect of this model uncertainty is within the bounds of uncertainty considered for the specific discharge uncertainty multiplier parameter used in the performance assessment. The NRC staff notes this is reasonable because having a flow path travel a longer horizontal distance in a particular unit will not significantly affect the flow rate, but if a permeable layer is vertically thicker or thinner than presumed in the model, the specific discharge rate would necessarily decrease or increase to accommodate the same volumetric flow.

The NRC staff evaluated DOE's consideration of model uncertainty related to the potential for a fault-dominated flow system with specific discharge focused in flow paths along major fault systems. This type of model conceptualization can produce rapid travel times by focusing high discharge rates into narrow zones (the fault systems) within the groundwater flow system. The NRC staff notes that this model uncertainty is appropriately addressed in the model abstraction by DOE's use of parameter ranges for effective porosity and specific discharge because the combination of very small values for effective porosity in tuff with high specific discharge rates introduces numerous realizations for the performance assessment that replicate the behavior of a fault-dominated flow and transport system. Numerous realizations in the performance assessment produce transport times for nonsorbing solutes on the order of 10–100 years for the glacial-transition climate state (SAR Section 2.3.9.3.4.1 and Figure 2.3.9-16). The NRC staff notes these realizations with rapid transport times reasonably represent the potential for focused high-permeability flow paths.

The NRC staff has evaluated the methods DOE used to characterize model uncertainty, and propagate the effects of this uncertainty, through the performance assessment abstraction and determines the methods are reasonable for the following reasons.

- DOE's performance assessment considered alternative conceptual models and modeling approaches to account for various uncertain FEPs.

- The modeling approach DOE used in the performance assessment is consistent with available data and current scientific understanding, and the results and limitations are appropriately considered in the abstraction.

- Several conceptual model uncertainties were defined and documented, and effects on conclusions regarding performance were properly assessed.

- Uncertainties in data interpretations for several aspects of the model were considered by analyzing reasonable alternative conceptual flow models that could not be ruled out by site data.

- Alternative modeling approaches DOE considered are consistent with available data and current scientific knowledge, and appropriately consider their results and limitations, using tests and analyses that are sensitive to the processes modeled.

2.2.1.3.8.3.5 Model Support Based on Comparison With Alternative Models or Other Information

In SAR Section 2.3.9.2.4, DOE presented its use of objective comparisons to build confidence in the saturated zone flow model abstraction. In this section, the NRC staff review focuses on support for the range of flow paths and specific discharge estimates considered for the saturated zone flow and transport model abstraction on the basis of relative importance to overall system performance.

The NRC staff reviewed information DOE used to support model-simulated groundwater flow paths. This information includes a comparison of simulated water-level elevations to those observed in wells not used in the model calibration (e.g., NC–EWDP Phase V data) (SAR Table 2.3.9-9). This comparison shows the largest differences between simulated and observed water levels generally occur in areas of steep hydraulic gradients near geologic features, such as the U.S. Highway 95 fault and the Solitario Canyon fault. Residual errors between observed and simulated water levels are generally smaller in the areas of simulated flow paths from the repository to the boundary with the accessible environment. The highest residual errors are generally in areas where water levels change by tens of meters (1 m = 3.28 ft) over short distances, and the overall range of residual errors is shown to be similar to the range of errors obtained during the site-scale saturated zone model calibration (SAR Figure 2.3.9-13). SNL (2007ax) described the calibration and confidence-building process. The NRC staff notes DOE's comparison of modeled results to water-level measurements from wells not used in the model calibration demonstrates the model reasonably reproduces present-day water levels in the model areas important to determining flow and transport paths from the repository to the boundary with the accessible environment. Although an independent analysis by the Center for Nuclear Waste Regulatory Analyses (Winterle, et al., 2003aa) showed that such residual error could be significantly reduced by adjusting the shapes of modeled geologic features, doing so does not appreciably change the modeled groundwater flow paths and specific discharge from the repository to the boundary with the accessible environment.

DOE discussed analyses by Freifeld, et al. (2006aa) of testing done in Nye County well 24PB that provides a range of estimates for specific discharge at the top of the Crater Flat tuff unit in the transition area from the volcanic aquifer to the valley-fill alluvial aquifer. On the basis of fluid electrical conductivity logging and distributed thermal perturbation sensor measurements, and assuming a porosity of 0.01, the estimated specific discharge in the flowing intervals ranged from 5–310 m/yr [16–1,018 ft/yr], as identified in Freifeld, et al. Section 3.3.3 and Table 4 (2006aa). The upper end of this range is significantly greater than the specific discharge rates considered in the performance assessment. DOE stated the high flow rate was observed in a relatively narrow interval of the borehole and that upscaling this estimate using an assumed median flow interval spacing of 25.8 m [84.6 ft] (from the parameter uncertainty distribution) reduces the estimated specific discharge to a range of 0.07–4.1 m/yr [0.2–13.5 ft/yr].

Freifeld, et al. (2006aa) proposed that "… additional data sets at other locations should be collected to examine whether the current data set is representative of the regional flow system near Yucca Mountain." The NRC staff notes, however, the observation of high-flow zones spaced tens of meters (1 m = 3.28 ft) apart is consistent with the conceptual model of flow in the fractured tuffs being through a network of relatively widely spaced fracture zones. The identified zones of high transmissivity are relatively thin and, when averaged over the entire penetrated thickness of the Crater Flat tuffs, which is appropriate for comparison to the model grid scale, the specific discharge estimates are reasonably consistent with the upper end of the range of uncertainty considered for specific discharge in the abstraction. Additionally, because the high groundwater flows entered the well at a lower interval and exited through an interval more than 40 m [130 ft] higher, the wellbore itself could be the cause of the high flow rates by connecting two vertically distinct permeable zones with groundwater flow driven by an upward vertical hydraulic gradient. Thus, the calculated high rates of specific discharge for this well could be more a reflection of flow driven by a local upward gradient that is short-circuited by the borehole and not representative of horizontal flow rates along groundwater flow paths in the aquifer system. Thus, the results reported in Freifeld, et al. (2006aa) are not conclusive regarding horizontal specific discharge at the scale of interest to the model abstraction, but generally support the concept of widely spaced, flowing intervals in the volcanic tuff units.

DOE also provided model support for estimates of potential future water table rise during future climates. This support included observations of mineralogical alteration and analysis of the strontium isotope ratio in calcite veins in the saturated and unsaturated rocks beneath the proposed repository footprint. This information is interpreted to indicate a past water table rise of approximately 85 m [278 ft] above present-day conditions beneath the proposed repository. Modeling studies using different assumptions about future recharge (Czarnecki, 1985aa; D'Agnese, et al., 1999aa) produce water-table-rise estimates from 130 to 150 m [427 to 492 ft] above present conditions beneath the repository. On the basis of this supporting information, DOE's performance assessment included a 120-m [394-ft] water table rise for the unsaturated zone flow and transport model abstraction. In the saturated zone flow and transport model abstraction, however, future water table water rise is not explicitly considered. Rather, DOE used an assumed rise of 85 m [278 ft] beneath the repository in an alternative conceptual model analysis to demonstrate the effect on saturated zone groundwater flow paths is not significant (see TER Section 2.2.1.3.8.3.4 for NRC staff review of this model uncertainty). South of the proposed repository near the RMEI location, the maximum extent of water table rise is constrained to approximately 30 m [98 ft] by the ground surface elevation and observed evaporite deposits at locations where springs flowed during past occurrences of elevated water table (SAR Section 2.3.9.2).

Observations from well UE-25p#1 cited by DOE as model support show the hydraulic head is 21 m [69 ft] higher in the carbonate aquifer system compared to the overlying volcanic aquifer system. The two aquifer systems are separated by a thick low-permeability aquitard. The calibrated site-scale saturated zone flow model reproduces this upward hydraulic gradient at well UE-25p#1, but with a lesser magnitude of a 6-m [20-ft] head difference across the aquitard unit. The NRC staff notes that, in addition to well UE-25p#1, the presence of this upward gradient along the area of model-simulated groundwater flow paths is also supported by observations in well NC–EWDP–2DB. DOE stated that these wells, which show an upward vertical gradient, are assigned a weight factor of 10 during model calibration to ensure the model will match the upward gradient. However, this statement appeared to be inconsistent with other information in the SAR where DOE apparently assigned a weight factor of 20 to UE-25 p#1 and a weight factor of 1 to Nye County well 2DB. In response to the staff's RAI, DOE confirmed the actual weight factors used in the model calibration were 10 for both well locations. DOE agreed to make changes in SAR Section 2.3.9.2.3.2 and the related supporting report (SNL, 2007ax) to reflect the actual values used.

The NRC staff has evaluated the approaches DOE used to compare performance assessment output to process-level model outputs and/or empirical studies and notes that the approaches are reasonable for the following reasons:

- The models implemented in DOE's performance assessment abstraction for saturated zone flow provided results consistent with output from detailed process-level models and empirical observations from field tests.

- Outputs of flow paths in the saturated zone abstractions reasonably reproduced the results of corresponding process-level models and empirical observations.

- The procedures DOE used to construct and test the mathematical and numerical models used to simulate flow paths in the saturated zone were well documented in the SAR and in supporting references (SNL, 2007an,aw,ax, 2008ab).

- The site-scale saturated zone flow model was developed on the basis of an underlying geologic framework, calibrated to minimize error compared to observed water levels, and compared to results from other models and field testing data not used in the model development (procedures that reflect reasonable and generally accepted scientific practices).

- DOE provided several supporting analyses to demonstrate the ranges of flow paths and specific discharge estimates used in the abstraction of flow paths in the saturated zone are consistent with site data and field tests.

2.2.1.3.8.4 NRC Staff Conclusions

The NRC staff notes that the DOE description of this model abstraction for flow paths in the saturated zone is consistent with guidance in the YMRP. The NRC staff also notes that the technical approach is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.8.5 References

BSC. 2005ak. "Saturated Zone Flow and Transport Model Abstraction." MDL–NBS–HS–000021. Rev. 03. AD 01, AD 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

CRWMS M&O. 1998ac. "Saturated Zone Flow and Transport Expert Elicitation Project." SL5X4AM3. Las Vegas, Nevada: CRWMS M&O.

Czarnecki, J.B. 1985aa. "Simulated Effects of Increased Recharge on the Groundwater Flow System of Yucca Mountain and Vicinity, Nevada-California." U.S. Geological Survey Water Resources Investigations Report 84-4344.

D'Agnese, F.A., G.M. O'Brien, C.C. Faunt, and C.A San Juan. 1999aa. "Simulated Effects of Climate Change on the Death Valley Regional Ground-Water Flow System, Nevada and California." U.S. Geological Survey Water-Resources Investigations Report 98-4041.

DOE. 2009an. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1.” Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Enclosures (7). Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bc. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.9.2.2 and 2.3.9.2.3), Safety Evaluation Report Volume 3, Chapter 2.2.1.3.8, Set 1.” Letter (January 30) J.R. Williams to J.H. Sulima (NRC). Enclosures (3): Numbers 1, 2, and 3. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008ab. DOE/RW–0573, “Safety Analysis Report Yucca Mountain Repository License Application.” Rev. 00. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Ferrill, D.A., J. Winterle, G. Wittmeyer, D. Sims, S. Colton, A. Armstrong, and A.P. Morris. 1999aa. “Stressed Rock Strains Groundwater at Yucca Mountain, Nevada.” *GSA Today*. Vol. 9, No. 5. pp. 1–8.

Freifeld, B., C. Doughty, and S. Finsterle. 2006aa. “Preliminary Estimates of Specific Discharge and Transport Velocities Near Borehole NC-EWDP-24PB.” LBNL–60740. Berkeley, California: Lawrence Berkeley National Laboratory.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. Washington, DC: NRC.

SNL. 2008ab. “Features, Events, and Processes for the Total System Performance Assessment: Analyses.” ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007an. “Hydrogeologic Framework Model for the Saturated Zone Site-Scale Flow and Transport Model.” MDL–NBS–HS–000024. Rev. 01. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aw. “Saturated Zone *In-Situ* Testing.” ANL–NBS–HS–000039. Rev. 02. ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ax. “Saturated Zone Site-Scale Flow Model.” MDL–NBS–HS–000011. Rev. 03. ACN 01, ERD 01, ERD 02. ERD 03. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ba. “Site-Scale Saturated Zone Transport.” MDL–NBS–HS–000010. Rev. 03. CAN 01, AD 001. Las Vegas, Nevada: Sandia National Laboratories.

Spitz, K. and J. Moreno. 1996aa. “A Practical Guide to Groundwater and Solute Transport Modeling.” New York City, New York: John Wiley & Sons.

Sun, A., R. Ritzi, and D. Sims. 2008aa. “Characterization and Modeling of Spatial Variability in a Complex Alluvial Aquifer: Implication on Solute Transport.” *Water Resources Research*. Vol. 44, W04402, doi:10.1029/2007WR006119.

Winterle, J.R. 2005aa. “Simulation of Spring Flows South of Yucca Mountain, Nevada, Following a Potential Future Water Table Rise.” ML052630242 San Antonio, Texas: CNWRA.
Winterle, J.R. and D.A. Farrell. 2002aa. “Hydrogeologic Properties of the Alluvial Basin Beneath Fortymile Wash and Amargosa Valley, Southern Nevada.” San Antonio, Texas: CNWRA.

Winterle, J.R., A. Claisse, and H.D. Arlt. 2003aa. “An Independent Site-Scale Groundwater Flow Model for Yucca Mountain.” Proceedings of the 10th International High-Level Radioactive Waste Management Conference (IHLRWM), Las Vegas, Nevada, March 30–April 2, 2003. La Grange Park, Illinois: American Nuclear Society. pp. 151–158.

(Intentionally Left Blank)

CHAPTER 12

2.2.1.3.9 Radionuclide Transport in the Saturated Zone

2.2.1.3.9.1 Introduction

This Technical Evaluation Report (TER) chapter addresses the U.S. Department of Energy's (DOE's) model abstraction[1] for transport of radionuclides in the saturated zone. DOE presented its description of this abstraction in Safety Analysis Report (SAR) Section 2.3.9 (DOE, 2008ab). Radionuclide transport in the saturated zone is a key process in the performance assessment for the proposed geologic repository at Yucca Mountain, Nevada. The performance assessment includes the flow of water from precipitation falling on Yucca Mountain, its migration as groundwater through the unsaturated zone above and below the repository, and the flow of groundwater in the saturated zone through the controlled environment to the accessible environment. This groundwater flow is the principal means by which radionuclides released from the repository are transported to the accessible environment. Because exposure to groundwater contaminated with radionuclides from the repository is one of the principal contributors to dose to the reasonably maximally exposed individual (RMEI), the performance assessment includes those components that affect significantly the timing and magnitude of transport for any radionuclides released from the repository. Radionuclide transport in the saturated zone, as described in SAR Section 2.3.9, includes the features, events, and processes (FEPs) that affect the movement of radionuclides from where they enter the saturated zone below the repository to the accessible environment boundary, approximately 18 km [11.18 mi] south of the repository, and their implementation (or abstraction) in the Total System Performance Assessment (TSPA).

In DOE's TSPA model, the saturated zone radionuclide transport abstraction receives information about the time-dependent flux of radionuclides released from the unsaturated zone to the water table below the repository. In turn, the saturated zone radionuclide transport abstraction provides information about the mass flux and arrival or transport times of radionuclides moving through the saturated zone to the representative volume in the accessible environment.

This chapter provides the U.S. Nuclear Regulatory Commission's (NRC's) evaluation of DOE's representation of radionuclide transport in the saturated zone as part of its performance assessment. Because DOE represented the barriers to radionuclide migration in the saturated zone in terms of radionuclide transport processes and how the processes were influenced by the natural features of the saturated zone, this chapter is organized by the major, risk-significant processes affecting transport.

2.2.1.3.9.2 Evaluation Criteria

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for the transport of radionuclides in the saturated zone, is guided by 10 CFR 63.114 (Requirements

---

[1]The term "abstraction" is defined as a representation of the essential components of a process model into a suitable form for use in a total system performance assessment. Model abstraction is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.

for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessment). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in TER Section 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The NRC staff review of the SAR and supporting information follows the guidance provided in the Yucca Mountain Review Plan (YMRP) Section 2.2.1.3.9 (NRC, 2003aa) Radionuclide Transport in the Saturated Zone, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of radionuclide transport in the saturated zone are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff’s review used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of the transport of radionuclides in the saturated zone important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as assessed by the NRC staff, are discussed in detail in this chapter. The NRC staff’s assessment is based both on risk information provided by DOE, and on NRC staff knowledge gained through experience and independent analyses.

2.2.1.3.9.3 Technical Evaluation

DOE provided information in SAR Section 2.3.9 and supporting documents that it uses to predict the transport of radionuclides in the saturated zone, from below the repository to the accessible environment. The NRC staff’s technical evaluation focuses on (i) how DOE represented the geological, hydrological, and geochemical features of the saturated zone in a framework for modeling the transport processes; (ii) how DOE integrated the saturated zone transport abstraction with other Total System Performance Assessment (TSPA) abstractions for performance assessment calculations; and (iii) how DOE included and supported important transport processes in process-level transport models and in the saturated zone radionuclide transport abstraction.

From a risk perspective, one of the most significant aspects of transport in the saturated zone is the extent to which the movement of contaminants to the accessible environment is limited or retarded. This performance aspect is most directly manifest in the calculated travel time for each radionuclide, given the transport processes and properties of the saturated zone. The calculated travel time results can be expressed as “breakthrough curves” for a given radionuclide or set of radionuclides that show the time history of transport along a given path. In terms of potential impact on repository performance, saturated zone transport most strongly affects those radionuclides which can decay appreciably over the time of transport; longer transport means that less of the radionuclide reaches the accessible environment. A radionuclide that is long-lived relative to its transport time, however, will reach the accessible environment with little decay. Even with retardation during transport, such a long-lived radionuclide will eventually “break through” the boundary with the accessible environment. Its flux will then approach a steady-state value, comparable to the flux of the radionuclide entering the saturated zone.

This simple transport concept is complicated by the existence of decay chains of radioactive daughters from long-lived parents released into the saturated zone (including, for example, decay chains from isotopes of uranium, thorium, neptunium, and plutonium). Decay of these radionuclides and their immediate daughters produces additional, shorter-lived radionuclides during transport. In a closed system, decay chains reach secular equilibrium where all component radionuclides have equal activity. Deviations from secular equilibrium, as either excess or deficient activity of a daughter relative to a parent, can develop in open or dynamic systems, particularly where parents and daughters have different chemical behavior. Given sufficient differences in chemical behavior, such disequilibria can manifest over time in groundwater transport systems, even where fluxes of long-lived parent radionuclides are in steady state. For the Yucca Mountain saturated zone transport system, where modeled release of contaminants is slow and limited (see performance assessment results discussed in TER Section 2.2.1.3.4), this is most apparent over longer performance periods, on the order of hundreds of thousands of years or more. The potential risk impact of disequilibria in decay chains on performance is considered in the subsequent sections of this technical evaluation;

specifically, TER Section 2.2.1.3.9.3.1, Section 2.2.1.3.9.3.1.2, Section 2.2.1.3.9.3.2.2, and Section 2.2.1.3.9.3.2,5.

2.2.1.3.9.3.1 Conceptual Model and Model Framework

DOE related Yucca Mountain site characteristics to a conceptual model of the saturated zone from beneath the repository to the accessible environment boundary in which the flow of water would transport released radionuclides through fractures, major faults, and porous alluvium. DOE used the same three-dimensional model grid for saturated zone transport modeling as for the site-scale saturated zone flow model (SAR Section 2.3.9.2; TER Section 2.2.1.3.8). There are two primary geological units through which water flows in the saturated zone: fractured volcanic tuff and alluvium. Unlike the unsaturated zone, where flow in the volcanic tuff occurs in fractures, faults, and matrix, DOE models flow in volcanic tuff of the saturated zone to occur only through fractures and faults. Water in the volcanic tuff matrix is connected to water in the fractures but is considered stagnant (flow velocities in the matrix are extremely low relative to the fast-flowing fractures).

DOE simulated the transport of radionuclides as dissolved species and as species sorbed to mobile, colloid-sized particles. These two modes of transport are subject to various physical and chemical processes that affect their transport in groundwater. DOE identified advection, dispersion, sorption and matrix diffusion, colloidal transport, and radionuclide decay and ingrowth as important transport-affecting processes, and incorporated these processes in the numerical models of radionuclide transport (SAR Table 2.3.9-1). DOE's conceptual model described how each of the transport-affecting processes influences the rate at which radionuclides travel through the saturated zone model relative to the rate that water travels (SAR Section 2.3.9.2). DOE used sensitivity analyses and single-realization analyses of TSPA simulations to demonstrate how the saturated zone transport abstraction integrated specific processes with the natural features of the saturated zone to slow the migration of radionuclides through the saturated zone, as detailed in SAR Section 2.1.2.3.6 and SNL Section 6.3.10 (2008ag).

The NRC staff notes that DOE included appropriate transport-affecting features, events, and processes (FEPs) in the conceptual model for saturated zone radionuclide transport and provided a technical basis for their inclusion in the conceptual model because DOE demonstrated these FEPs affect transport using appropriately designed laboratory and field tests, and analog data. The NRC staff evaluates DOE's technical bases for excluding other transport-affecting FEPs in TER Section 2.2.1.2.1 and notes that DOE's explanations for excluding the identified FEPs are adequate because they were supported by site characterization data, field tests, laboratory experiments, and natural analogs. One specific excluded FEP—the effect of temporal changes in groundwater composition on radionuclide concentrations in the saturated zone—is examined in more detail in TER Section 2.2.1.2.1, because information exists that suggests temporal changes occur (Turner and Pabalan, 1999aa), and these changes could result in releases of certain radionuclides sorbed over time to the saturated zone groundwater.

Turner and Pabalan (1999aa) offer alternative reasons for the observations of apparent temporal variability other than by natural causes. These include sampling from different flow zones in the same well, differences in sampling, sample preparation techniques, filtration, and pumping times. The NRC staff notes that this process, although likely to occur, would not affect radionuclide concentrations significantly if the probabilistic analyses include temporal uncertainty.

On the basis of field and modeling results, DOE determined groundwater flow and migration of radionuclides in the saturated zone would begin in fractured volcanic rock beneath the repository and travel southeasterly toward Fortymile Wash before turning in a southerly direction beneath the wash to the accessible environment, approximately 18 km [11.18 mi] south of the repository. About 10 km [6.21 mi] south of the repository along this path, the water and radionuclides are expected to flow from the volcanic rock and enter the alluvium. The location of the transition from volcanic rock to alluvium is a key geologic data uncertainty in DOE's transport abstraction (BSC, 2005ak). This uncertainty was incorporated in DOE's model as an alluvium uncertainty zone, which was constrained by geologic data from Nye County Early Warning Drilling Program (NC–EWDP) wells. The length of travel of radionuclides in the volcanic rock or alluvium is dependent on the source zone (location of release from the repository), the value of horizontal anisotropy in permeability, and the size of the alluvium uncertainty zone.

In DOE's model, the two geologic media encountered in the saturated zone, fractured volcanic rock and alluvium, are expected to have very different effects on water flow and radionuclide transport. The fractured volcanic rock has low porosity (void space) and sparsely distributed fractures that control flow and transport. Consequently, the water and radionuclides move relatively quickly through fractures in the volcanic rock. The sparse distribution of fractures limits interactions between the radionuclides and the rock that could slow radionuclide transport. In contrast, the alluvium was modeled as a porous medium, with significantly higher porosity than the volcanic rock. Consequently, water moves more slowly through the alluvium. Because of the way in which the alluvium was deposited, some preferential flow paths could exist in buried gravel deposits. DOE accounted for these potential higher velocity flow paths by including effective porosity as an uncertain parameter, with a range of values to accommodate these features in the model abstraction (BSC, 2005ak). Consistent with results from field-based transport testing (SNL, 2007aw), DOE did not take credit for one of the retardation processes, matrix diffusion, that might result from the dual porosity aspect of the alluvial system.

DOE modeled radionuclide release from the unsaturated zone into the saturated zone as occurring at one to four point sources near the water table. The point source locations were randomly sampled from within each of the four corresponding unsaturated zone source regions that generally represented preferential flow pathways in the unsaturated zone flow model.

In DOE's TSPA model, two model abstractions of saturated zone flow and transport were implemented. The primary model of radionuclide transport in the saturated zone was a site-scale, three-dimensional, single (effective)-continuum, dual-porosity, particle-tracking transport model. The dual-porosity aspect allowed for consideration of matrix diffusion in fractured volcanic tuff. The two porosities refer to those of the fractures and of the matrix, while the effective continuum aspect of the approach allowed DOE to assign average values to flow and transport parameters applied to cells of the numerical model representing the system.

Radionuclides whose transport was simulated through the saturated zone were subdivided into 12 groups on the basis of their similar transport characteristics (primarily based on sorption behavior, as discussed in detail in TER Sections 2.2.1.3.9.3.2.2 and 2.2.1.3.9.3.2.4). Radionuclides with half-lives of less than 29 years were not explicitly transported in DOE's saturated zone flow and transport abstraction. Instead, for shorter-lived radionuclides that are part of decay chains with long-lived parents (e.g., U-238 series), DOE determined daughter concentrations in groundwater using an assumption of secular equilibrium. Secular equilibrium occurs when the activity of a short-lived daughter radionuclide is equal to that of a longer-lived parent radionuclide, due to radioactive ingrowth. Differences in sorption behavior between parent and daughter radionuclides can affect their relative concentrations in groundwater, especially at local scales, such as near the boundary of the controlled area where water is pumped from wells. The NRC staff notes that the DOE assumption of secular equilibrium may not be valid for some parent-daughter pairs and can result in uncertainties not included in the conceptual model. Evaluation of the impact of this assumption on calculations of repository performance is detailed in TER Section 2.2.1.3.9.3.1.2, Section 2.2.1.3.9.3.2.2, and Section 2.2.1.3.9.3.2.5.

DOE's three-dimensional model abstraction used flow fields and other hydrologic characteristics defined by the three-dimensional saturated zone flow model and calculated unit breakthrough curves for each of the 12 radionuclide groups transported in the saturated zone. The three-dimensional model was run, and the unit breakthrough curves were developed and stored external to the TSPA model, as outlined in SNL Section 6.3.10 (2008ag).

The saturated zone radionuclide transport abstraction was coupled to input from the unsaturated zone and output to the biosphere using the convolution integral method. In this method, a unit saturated-zone radionuclide mass breakthrough curve was computed (by the three-dimensional model) for a step-function mass flux source. This breakthrough curve was then combined with the radionuclide mass flux history from the unsaturated zone to produce a radionuclide mass flux history that was output to the biosphere. Within TSPA, the convolution integral technique was implemented by a module called *SZ_Convolute*. *SZ_Convolute* was also used to apply changes in specific discharge due to climate change and to correct radionuclide releases from the three-dimensional model for the effects of radioactive decay. The convolution integral method rests on the key assumptions of linear behavior and steady-state saturated zone flow conditions. The saturated zone output mass that crosses the accessible boundary in a year was assumed dissolved in 3.7×10^9 L [3,000 acre-ft] of water.

DOE considered three climate states in modeling the initial 10,000 years after repository closure: (i) current, (ii) monsoonal, and (iii) glacial transition conditions. For evaluating repository performance after 10,000 years following repository closure, DOE used "constant-in-time" climate with a prescribed deep percolation rate. The saturated zone transport model abstraction assumed steady-state flow conditions under each specified climate state and assumed the steady-state conditions were achieved instantaneously for each climate transition. Although potential water table elevation changes and flow path changes could occur for different climate conditions, DOE did not implement these changes explicitly in the saturated zone transport abstraction. Instead, DOE used specific discharge multipliers to increase flux for the wetter future climates. Also, the potential change in water table elevation near the repository was simulated by shortening the flow path length through the unsaturated zone during wetter climates, and releasing instantaneously to the saturated zone any radionuclides that are present in the portion of the unsaturated zone that is inundated at the time of climate change.

The NRC evaluates the use of specific discharge multipliers in TER Section 2.2.1.3.8.3.3. The specific discharge multipliers affect radionuclide transport. Increasing the specific discharge increases the radionuclide mass that reaches the accessible environment boundary within the 10,000 year period. However, DOE demonstrated that the range of specific discharges does not affect the retardation processes considered in this chapter. DOE evaluated the effects of changes in water table elevations and flow due to climate change by using the three-dimensional site-scale transport model to generate particle tracks, as described in SNL Appendix E (2007ba). DOE stated that the simulation results showed that the path lengths and travel times for radionuclides increased relative to the model formulation used in performance assessment, and thus, exclusion of these changes in the saturated zone abstraction model did not result in an underestimation of dose, as outlined in SNL Appendix E (2007ba). In its response to the NRC staff request for additional information concerning geochemical assumptions of the saturated zone transport models under different climatic conditions (DOE, 2009df), DOE described how elevated water-table flow paths would pass through rocks containing higher percentages of zeolites. In general, water movement and transport of certain radionuclides is slower through rocks containing zeolites compared to non-zeolitic rocks. Instead of explicitly considering these different pathways, DOE chose to simplify its climate change model by employing only the specific discharge multipliers (see TER Section 2.2.1.3.8.3.3). The NRC staff notes that DOE's climate change abstraction is reasonable, because TSPA model results suggest that unretarded radionuclides, which are relatively unaffected by changes in chemical conditions, will dominate the radionuclide release to the environment in the initial 10,000 years after repository closure, when changes in climate are considered.

The second saturated zone model abstraction was a one-dimensional transport model. The main purpose of the one-dimensional model was to calculate the radioactive decay, ingrowth, and transport for second-generation daughter radionuclides for four decay chains—the actinium, uranium, thorium, and neptunium series. The one-dimensional transport model was implemented as four groups of GoldSim (GoldSim Technology Group, 2006aa) pipe elements. One group was used for each of the four repository source regions. Each group of pipe elements consisted of three segments, representing the volcanic tuffs (the first two segments) and the alluvium (the last segment). The lengths of the last two segments were considered uncertain and were derived from particle-tracking results of the three-dimensional saturated zone model. Groundwater-specific discharge values in each pipe segment were also estimated from the three-dimensional model given in SNL Section 6.3.10 (2008ag). Where possible, the one-dimensional model used the same transport parameters, such as sorption coefficients, as the three-dimensional model. DOE adjusted other features of the one-dimensional model, such as dispersivity and flow tube diameters, to improve consistency between the breakthrough curve results from the two abstractions. The NRC staff notes that this method of adjusting some parameters is reasonable. The parameters that were adjusted were shown to be unimportant to performance. Those parameters that were important to performance were held the same for the two models. Furthermore, the use of two approaches to simulate transport is consistent with YMRP guidance, which describes how alternative modeling approaches consistent with available data can be used to support model abstractions.

The NRC staff notes that DOE provided a technical basis for all features and conditions of the saturated zone transport model framework, consistent with the guidance in the YMRP, with the exception of the assumption of secular equilibrium for certain decay products to which the receptor can be exposed (staff evaluation of this assumption is in TER Section 2.2.1.3.9.3.1.2, Section 2.2.1.3.9.3.2.2, and Section 2.2.1.3.9.3.2.5). This is based on the NRC staff's understanding of the Yucca Mountain natural system, obtained from field observations and independent analysis of saturated zone transport processes, and DOE's proposed repository design. The NRC staff notes that DOE included important Yucca Mountain site characteristics that are expected to affect radionuclide transport, such as flow in the fractures of the volcanic tuff, the porous nature and uncertain extent of the alluvium, and uncertainties in groundwater flow rates. DOE used Yucca Mountain site characterization data to assign geologic, hydrologic, and radionuclide transport parameter values to specific rock units or to define ranges of values for these properties to address uncertainty about the natural variability of the system. For short-lived decay-chain radionuclides (see discussion of relevant radionuclides in TER Chapter 2.2.1.3.14, Biosphere Characteristics), DOE, in its Safety Analysis Report, assumed that their concentrations in groundwater at the boundary of the controlled area could be determined through secular equilibrium with their dissolved long-lived parents. In some cases this assumption can lead to an underestimate of the amount of radionuclides present in the contaminant plume in the accessible environment. DOE provided further analysis of this topic in its response to an NRC staff request for additional information. As previously noted, the NRC staff evaluation of these specific analyses is discussed in TER Section 2.2.1.3.9.3.1.2, Section 2.2.1.3.9.3.2.2, and Section 2.2.1.3.9.3.2.5.

2.2.1.3.9.3.1.1 Model Integration for TSPA

DOE's saturated zone transport abstraction simulated the transport of dissolved radionuclides and colloid-associated radionuclides through the saturated zone from beneath the repository, generating breakthrough curves at the accessible environment boundary for 27 radionuclides that DOE determined were risk significant. For TSPA calculations, DOE integrated the saturated zone radionuclide transport abstraction with three other model components: the unsaturated zone transport abstraction model (SAR Section 2.3.8), the site-scale saturated zone flow model (SAR Section 2.3.9), and the biosphere model (SAR Section 2.3.10).

2.2.1.3.9.3.1.2 Unsaturated Zone/Saturated Zone Boundary Condition

In the unsaturated zone transport abstraction (SAR Section 2.3.8), radionuclides released from waste packages migrated through the fractures and rock matrix at rates affected by flow fields generated from the site-scale unsaturated zone flow model. The boundary through which radionuclides from the unsaturated zone passed to the saturated zone was divided into four regions (or subareas). Releases of radionuclides that occurred within any portion of each of the subareas were collected for release into the saturated zone at one point (also within the same subarea). Each point source mass of radionuclides was transferred to the three-dimensional (via *SZ_Convolute*) and the one-dimensional saturated zone transport model abstractions. The three-dimensional model breakthrough curves were generated by randomly sampling different locations within each of the four subareas to create a starting point for the saturated zone flow path(s). In contrast, the one-dimensional model uses a fixed, centroid location within each subarea as a starting point for each of its four flow tubes. As a result of the different starting points for the three-dimensional and one-dimensional simulations, the path lengths were not necessarily the same for both methods. DOE described the comparison of the three-dimensional and one-dimensional simulations as not consistently overestimating or underestimating the travel time to the accessible environment. The NRC staff notes that this description characterizes the uncertainty associated with the methods. The radionuclide species and its type of transport remained consistent in the transfer of mass from the unsaturated zone to the saturated zone. For example, radionuclides associated with irreversible colloids were passed as irreversible colloids, while those associated with reversible colloid transport were repartitioned according to the different colloid concentrations encountered in the saturated zone. All radionuclide mass released from the unsaturated zone was transferred to flowing fractures in the saturated transport models.

DOE sensitivity analyses of the effects of releasing radionuclide mass from the unsaturated zone as point sources indicated that the point source releases generally produced faster breakthroughs, as described in SNL Section 6.8.4 (2007ba). DOE also noted that the release of radionuclides as a point source generally produced a plume with less dispersion.

The NRC staff notes that DOE reasonably transferred radionuclide mass between the unsaturated and saturated zone models. The NRC staff also notes that DOE applied consistent conceptual models and estimates of data and model uncertainty for site conditions relevant to the unsaturated and saturated transport abstractions.

Saturated Zone—Accessible Environment Boundary Condition

The saturated zone transport model passes to the biosphere model the time-varying mass of radionuclides that cross the boundary with the accessible environment. The saturated zone transport model explicitly simulates the transport of 27 radionuclides. These radionuclides are carried in the groundwater in the dissolved state and as temporarily and/or permanently associated with colloids. Parameters affecting the transport in the saturated zone (see TER Sections 2.2.1.3.9.3.2.2 and 2.2.1.3.9.3.2.4) were assigned to the 27 radionuclides. In the TSPA model four additional radionuclides—Ac-227, Ra-228, Th-228, and Pb-210—were assumed to be in secular equilibrium and were released to the biosphere at the same activity as the activity in ground water of their long-lived parents, Pa-231, Th-232, U-232, and Ra-226, respectively. In response to a request for additional information that asked for justification for setting the released activity of daughters equal to that of their released parents, DOE (2009de) acknowledged that the activities of these daughters are affected by differences in the sorption characteristics of the parents and daughters. DOE (2009de) analyses indicated that mean activities for the daughters of Th-232/Ra-228, Pa-231/Ac-227, Ra-226/Rn-222, and Ra-226/Pb-210 parent-daughter pairs could range from 1 (e.g., Ra-226/Pb-210) to more than 1,400 (e.g., Ra-226/Rn-222) times that of the respective activities of the parents. DOE defined the sorption enhancement factor (SEF) as the ratio of the daughter activity relative to the activity of its parent in groundwater, resulting from the differences in sorption coefficients of the parent and daughter. DOE concluded that explicitly including the increased daughter activities in the dose calculation of the four parent/daughter pairs would increase the calculated maximum total mean annual dose to the RMEI by 20 percent, from 0.02 to 0.024 mSv [2.0 to 2.4 mrem]. The NRC staff notes that although the method used in DOE (2009de) was reasonable to address the effect of the secular equilibrium assumption, the K_d distributions used in DOE's response were biased to lower values, potentially decreasing the magnitude of the effect (see the detailed discussion in TER Section 2.2.1.3.9.3.2.5).

The NRC staff performed independent sensitivity analyses (Bradbury, 2010aa) to determine the effect of sorption on concentrations of other decay-chain radionuclides to which the receptor can be exposed. Whereas DOE assumed all radionuclides with half-lives of less than 29 years were in secular equilibrium with their parents, the NRC staff explicitly included in its transport analyses radionuclides with half-lives as short as 1 hour. These shorter lived radionuclides were included to consider possible effects for direct ingestion of contaminated drinking water from wells in the groundwater at the location in the accessible environment that includes the highest concentration of radionuclides in the plume of contamination. The independent analyses used pipe elements of the GoldSim modeling program to simulate one-dimensional transport with sorption and radioactive decay and ingrowth for the four radioactive decay series: actinium, neptunium, thorium, and uranium. The NRC staff's analyses showed that some daughter activities were less dependent on the relative K_d of the immediate parent than on the relative K_d of the distant parent that acts as the source of radioactivity to the pipe. For example, in a simple uranium decay series, a constant source of U-234 injected into the pipe will decay as it is transported down the pipe producing Ra-226, which subsequently decays to Rn-222. Eventually, a steady state is approached in which the fluxes of U-234, Ra-226, and Rn-222 remain relatively constant. The flux of Rn-222 at this point depends predominantly on the K_d of U-234 and is largely independent of the K_d of Ra-226. Consequently, simulations that involve changes in the K_d of Ra-226 but keep the

U-234 K_d and source flux unchanged result in the same Rn-222 flux. This suggests that K_ds of radionuclides other than those identified in DOE's response to the request for additional information could affect concentrations of short-lived radionuclides. The NRC staff's analyses showed that, assuming sufficient difference in sorption between U-234 and one of its decay products, Po-210 (half-life of 138 days), measurable excess activity of Po-210 can develop in the plume at the boundary of the accessible environment. DOE did not analyze transport of Po-210 and thus did not provide values for the sorption coefficient of this nuclide. The NRC staff evaluation of the specific geochemical conditions necessary to produce significant excess Po-210 in groundwater, and the potential for this to occur in the Yucca Mountain system, is in TER Section 2.2.1.3.9.3.2.5.

The saturated zone transport model passes time-dependent fluxes of the 27 radionuclides whose transport is explicitly simulated in TSPA, plus the four radionuclides that are assumed to be in secular equilibrium with their immediate parents. In turn, the biosphere model calculates biosphere dose conversion factors for the groundwater exposure scenario equivalent to the annual dose from all potential exposure pathways that the RMEI would experience as a result of the release of a unit concentration of 1 Bq/m^3 [0.227 dpm/gal] of the primary radionuclide in groundwater at the accessible environment boundary. The biosphere model considers 75 radionuclides composed of the 31 primary radionuclides and their relatively short-lived daughters, which are assumed to be in secular equilibrium (i.e., have the same activity concentrations as those of their parents).

The NRC staff notes that DOE's assumption of secular equilibrium adds uncertainty to the coupling of the biosphere model to the saturated zone transport model. Although DOE (2009de) described the effects of differences in sorption characteristics on four pairs of parent-daughter radionuclides, other pairs exist for which the assumption of secular equilibrium may not be supported. Independent NRC staff analyses indicate calculated fluxes could change for short-lived daughters if they were explicitly considered in the transport model. The magnitude of the potential effect of these nuclides on dose depends on their dose coefficients as well as their excess activity in the groundwater. The added uncertainty resulting from not including these nuclides in the transport model is reasonable given that it does not significantly increase the mean of the expected annual dose to the reasonably maximally exposed individual.

2.2.1.3.9.3.2 Saturated Zone Transport Processes

In DOE's saturated zone transport abstraction, the migration of radionuclides through the saturated zone is influenced by the transport-affecting processes of advection and dispersion, sorption, matrix diffusion, and colloid-facilitated transport, as well as radioactive decay and ingrowth (SAR Section 2.3.9.1). NRC staff review of these processes focuses on the system description and integration, data support, data uncertainty, and model uncertainty and support.

2.2.1.3.9.3.2.1 Advection and Dispersion

Advection is the process by which radionuclides, both dissolved and associated with colloids, are carried in flowing water. Overall, DOE considered advection to be the most important transport process in the saturated zone because, as DOE stated, advection is the primary mechanism driving the migration of radionuclides in the saturated zone, as outlined in BSC Section 6.3.1 (2005ak). Accordingly, the uncertainty of specific discharge, or the measure of flow in the saturated zone, has the greatest effect on travel times, as described in SNL Section 6.8.4 (2007ba).

Unlike DOE's model framework for unsaturated zone radionuclide transport, which allows radionuclides to travel advectively in both fractures and the matrix, in the saturated zone, radionuclides in the fractured volcanic tuff move advectively only in the fractures and fault zones. Radionuclides can, however, move into and out of the matrix by diffusion, driven by concentration gradients (see TER Section 2.2.1.3.9.3.2.3). Hydrologic testing in boreholes in the volcanic aquifer revealed that flow through fractures was generally spaced at intervals significantly greater than the spacing of the fractures themselves as determined in drill core logging. This site characteristic was included in the model abstraction as the uncertain parameter *flowing interval spacing in volcanic units*. Along with this parameter, DOE coupled an uncertain parameter, termed *Fracture porosity in volcanic units* (SAR p. 2.3.9-132)*, or flowing interval porosity in the fractured tuffs* (SAR Section 2.3.9.3.2.1). Values for this parameter were estimated using various conservative tracers and reactive tracers in C-Wells Complex testing (SNL, 2007aw). DOE used a combination of the spacing and the porosity parameters to describe the characteristic of preferential pathways through the fracture volcanic aquifer.

The NRC staff notes that the methods to constrain the values of these parameters are reasonable because these methods are supported by site characteristic data and field testing. Although a limited number of locations have been characterized to determine porosity and spacing, DOE's estimation of the distribution range (large spacing and small porosity) compensates for the uncertainty in these parameters.

On the basis of information DOE provided about saturated zone radionuclide transport by advection in SAR Section 2.3.9.3 and references therein, NRC staff notes that advective radionuclide transport in DOE's transport abstraction depends on the flow field information DOE's site-scale saturated zone flow model supplies to the transport abstraction. The NRC staff notes that DOE's implementation of advective radionuclide transport is integrated with the site-scale saturated zone flow model, using the same three-dimensional model grid, hydrologic properties, modeling approach, and flow fields to represent advection fluxes. The NRC staff's technical evaluation of DOE's site-scale saturated zone flow model is provided in TER Section 2.2.1.3.8.

Advection in alluvium, like advection in the volcanic aquifer, involves preferential pathways. In the alluvium, DOE used an effective porosity parameter to compensate for the potentially reduced volume of the alluvium through which flow might occur. A smaller effective porosity results in higher average linear velocity (i.e., the distance water moves through porous material per unit time). DOE modeled the alluvium as a single-continuum medium. Consequently, there is no exchange of water or radionuclides between the effective porosity (where flow occurs) and the rest of the porosity (where no flow occurs) modeled in the alluvium. Although this conceptual model does not explicitly account for possible flow paths within the alluvium, the NRC staff considers the approach to be a reasonable method for incorporating preferential pathways and minimizing the travel time through the alluvium because it is compatible with site characteristics, such as the occurrence of gravel paleochannels and lenses of clay. Field evidence of tracer exchange between effective porosity, where water is flowing, and porosity, where water is stagnant, was inconclusive (SNL, 2007aw). The NRC staff considers DOE's approach to simulating advection in the alluvium is reasonable because the approach includes processes and conditions shown to be present in the alluvium and excludes processes that have not been shown to occur.

DOE provided supporting information about advective transport processes from saturated zone tracer experiments in densely welded, fractured tuffs (fracture-flow dominated systems) at the C-Wells Complex and in the porous alluvium (SAR Section 2.3.9.3.2.1; SNL, 2007aw). The NRC staff notes that DOE provided a technical basis for implementing advective radionuclide transport in the saturated zone radionuclide transport abstraction supported by the field testing in the volcanic and alluvial aquifers.

Dispersion describes the transverse spreading, perpendicular to flow, both horizontal and vertical, and longitudinal spreading, parallel to flow, of dissolved radionuclides in response to localized differences in flow conditions. At the large scale of the saturated zone transport model grid framework, DOE considered the effect of dispersion to be minimal (SAR Section 2.3.9.3.2.1). However, to allow transport calculations to provide an analysis of radionuclide travel time distributions, DOE included a longitudinal dispersion term in the transport model to capture the arrival of a dispersed solute front at the accessible environment boundary (SAR Section 2.3.9.3.2.1). Inclusion of longitudinal dispersion is supported by the field evidence of preferential pathways in the saturated zone in the vicinity of Yucca Mountain. The NRC staff examined DOE's assumptions and technical approach for including dispersion as a transport process in the saturated zone transport abstraction. On the basis of the NRC staff's recognition that dispersion does not significantly affect the results of transport calculations in a slowly evolving system where radionuclide concentrations gradually increase with time, the NRC staff notes the approach is reasonable.

2.2.1.3.9.3.2.2 Sorption

As discussed previously in TER Section 2.2.1.3.7.3.2.2, sorption is a general term for chemical and physical processes that transfer a fraction of dissolved species to the surface of a solid phase. Depending on specific properties of the dissolved species, the solid, and the liquid, some radionuclides will sorb to the solid, some will sorb more onto solids than others, and some will not sorb at all. Specific sorption processes DOE considered include ion exchange reactions, in which ions of one element replace ions of another element within a mineral structure, and surface complexation, which involves reactions that form bound species at the mineral–water interface. As modeled in the DOE TSPA, sorption of radionuclides onto the fractured volcanic tuff matrix or alluvium results in retardation, or slowing, of transport relative to rates of water flow through the saturated zone. In contrast, radionuclide sorption onto mobile colloids may enhance the transport rate relative to radionuclides that attach onto a stationary solid (see TER Section 2.2.1.3.9.3.2.4). Sorption is an important process contributing to the barrier capability of the saturated zone (SAR Section 2.3.9). In particular, sorption within the alluvium effectively delays the transport of moderately and strongly sorbing radionuclides for thousands of years or longer (SAR Sections 2.3.9 and 2.1.2.3.6). Sorption of dissolved thorium, americium, and protactinium is so effective at slowing their movement, that on entering the saturated zone, these radionuclides cannot reach the accessible environment within 1 million years. To be present at the accessible environment boundary, these radionuclides either are transported through the saturated zone as colloids or are ingrown as decay products of mobile parents. DOE noted that the primary controls on sorption are (i) characteristics of the minerals surfaces onto which sorption occurs, (ii) chemistry of groundwater in the saturated zone, and (iii) sorption characteristics of each element (SAR Section 2.3.9.3.2.2).

DOE represented sorption in the saturated zone with a sorption coefficient (K_d), an empirically determined or modeled value that represents the ratio of the sorbed-phase radionuclide concentration to the dissolved-phase radionuclide concentration. Low values of K_d indicate that little or no sorption (i.e., $K_d = 0$) occurs; higher values indicate moderate or strong sorption, and therefore retardation. Retardation by sorption is expressed in transport calculations by a retardation factor (R_f) that depends on the K_d value and the physical properties (porosity and density) of the solid medium through which the radionuclide is transported. Thus, a retardation factor (R_f) equal to one indicates the solute is transported at the velocity of groundwater, while an $R_f > 1$ indicates the solute transport is delayed relative to groundwater. Retardation calculations assume that K_d does not vary with changes in radionuclide concentration, sorption and desorption reactions are fast relative to the flow rate, and bulk chemical composition of the groundwater is constant (Davis and Curtis, 2003aa; Langmuir, 1997aa; Davis and Kent, 1990aa). DOE provided information, satisfactory to the staff, to support each of these assumptions (DOE, 2008ab; BSC, 2005ak; SNL, 2007aw,ba). For example, DOE varied the concentration of radionuclides in sorption experiments to determine the effect on K_d. DOE varied the duration of sorption/desorption experiments to determine the rate of these reactions. By using groundwaters of different compositions, DOE demonstrated the effect of bulk chemistry on K_ds.

DOE assumed that sorption of dissolved radionuclides would occur only in the matrix of the volcanic tuff or in the alluvium; dissolved radionuclides transported in fractures do not sorb to fracture surfaces in the DOE model (SAR Section 2.3.9.3.2.2). DOE excluded sorption onto fracture surfaces because of high uncertainties in the nature of fracture coatings (BSC, 2005ak). However, solutes transported through designated fault or fracture zones could undergo sorption depending on the characteristics of the fault or fracture zone (BSC, 2005ak). In fracture zones, a small portion of the rock matrix within the fracture zone was conceptualized as having rapid diffusion and a retardation factor was calculated accordingly (BSC, 2005ak). In contrast to dissolved radionuclides, mobile colloids were retarded within fractures of the volcanic tuff (SAR Section 2.3.9.3.3). Laboratory and field-scale experimental results supported DOE’s conceptual model of colloid retardation in fractures (BSC, 2005ak; SNL, 2007aw).

In the saturated zone transport model and abstraction, DOE assumed that four radioelements (carbon, chlorine, iodine, and technetium) were nonsorbing and assigned a fixed value of $K_d = 0$ (corresponding to $R_f = 1$) for each. While results of field-based testing in the alluvium indicated that the transport of the important radioelements, technetium and iodine, may be retarded, laboratory-based sorption tests indicated a $K_d = 0$ was warranted, as described in SNL Appendix G (2007ba). For the remaining radioelements modeled in saturated zone transport calculations (americium, cesium, neptunium, plutonium, protactinium, radium, selenium, strontium, thorium, tin, and uranium), DOE developed ranges and statistical distributions of K_d values for each radioelement and for each modeled rock unit from a combination of empirical data, process modeling, statistical analyses, and expert judgment, as shown in SAR Table 2.3.9-4 and SNL Appendices A, C, G, and J (2007ba). The NRC staff notes that the methods applied to determine distributions of K_ds are consistent with YMRP guidance because DOE followed reasonable practices for determining sorption.

For sorption modeling, DOE grouped the various stratigraphic units in the saturated zone into two geologic media that have different sorption characteristics: fractured volcanic tuff and alluvium (SAR Section 2.3.9.3.2.2). DOE measured sorption data from batch and column experiments that used site-specific crushed tuff samples and alluvium and used saturated zone water samples from wells in the saturated volcanic tuff (UE-25 J-13), carbonate aquifer (UE-25 p#1), and alluvium (various EWDP wells). Water from wells UE-25 J-13 and UE-25 p#1 were used for batch experiments with volcanic tuff samples, while experiments with alluvium utilized water from the alluvial aquifer, as described in SNL Appendix G (2007ba). DOE argued that these water chemistries bracket the major ion chemistry observed in the saturated zone, as outlined in BSC Appendix A (2005ak). For the long-lived actinides (americium, neptunium, plutonium, and uranium), DOE further characterized the effects of variability in geochemistry and mineral surface area using a non-electrostatic surface complexation modeling approach, supported by Davis, et al. (1998aa) and BSC Appendix A (2005ak). In some cases, DOE supplemented the experimental and modeling sorption data with data from the open literature (BSC, 2005ak). In TSPA, DOE sampled K_d values from the specified ranges to account for experimental uncertainty and variability in geologic conditions, including water chemistry and rock type, as shown in SAR Table 2.3.9-4; BSC (2005ak); SNL Appendices A, C, G, and J (2007ba); and DOE Enclosure 3 (2009am).

The NRC staff is familiar with geochemical controls on radionuclide sorption and with the experimental data DOE used to develop the TSPA K_d distributions. The NRC staff notes that the major ion chemistry (e.g., calcium, sodium, bicarbonate) of the waters used in DOE sorption experiments is comparable to that of saturated zone waters, as described in BSC Appendix A (2005ak). The UE-25 J-13 and UE-25 p#1 water chemistries bound the ranges reported for saturated zone water chemistries for major ions such as sodium, calcium, and bicarbonate, and other parameters such as pH and redox state. These chemical characteristics are likely to be the most important for radionuclide sorption (e.g., Turner and Pabalan,1999aa). The NRC staff further notes that the chemistries of alluvial aquifer waters used in alluvium sorption experiments are representative of conditions in the alluvium. DOE also used surface complexation modeling, similar to independent models the NRC staff developed (e.g., Turner, et al., 2002aa), to examine the effects of broader chemistry ranges on several radionuclides. The NRC staff notes that DOE's use of surface complexation modeling to extend the limited chemical conditions in the batch crushed tuff experiments strengthened the technical basis for the upper and lower limits of sorption coefficients for the targeted actinides.

DOE identified mineral surface area and particle size as potential sources of data uncertainty related to the use of crushed tuff and alluvium in experiments. DOE referenced studies both from within and outside the DOE program that indicate the effects of particle size on sorption are typically small except for the very fine (e.g., clay-sized) fraction (BSC, 2005ak). The smallest particle size results in higher K_ds. The general DOE approach to addressing this uncertainty was to use batch experiments for a range of particle sizes and to bias the minimum and maximum limits for the K_d distributions toward lower (weaker sorption) values, as shown in DOE Table 1.1.2-1 (2009am). SNL Appendix A (2007ba) gave the range of K_ds used in the TSPA. In SNL Appendix J (2007ba), however, DOE described how batch sorption experiments may underestimate the K_d that should be applied to transport in the field. The experiments described in SNL Appendix J (2007ba) involved sorption/desorption of uranium and neptunium. The effective K_ds for these radionuclides were up to two orders of magnitude greater than those used in the TSPA. DOE stated that the higher K_ds are likely due to multiple sorption sites of different strengths. There are strongly sorbing sites and weakly sorbing sites on the solids in the saturated zone. Batch sorption experiments are of short duration and preferentially measure the weak sites. Desorption experiments of longer duration measure the reactions with the stronger sites. DOE chose not to use the effective K_ds from SNL Appendix J (2007ba) for uranium and neptunium, because they would be inconsistent with the K_ds of the other radionuclides determined by the batch sorption method.

In selecting experimental data to inform the TSPA K_d distributions, DOE excluded data from experiments where the final radionuclide concentration indicates that the solubility limit of the radionuclide may have been exceeded, as described in SAR Section 2.3.9.3.2, BSC Appendix A (2005ak), and SNL (2007ah). However, the data DOE provided appear to indicate that some experiments were conducted at initial concentrations above the solubility limits for some radionuclides, as described in BSC Appendix A (2005ak). The results from these experiments could be biased by precipitation effects, although the range and magnitude of the sorption coefficients appear to be consistent with experiments conducted using lower, undersaturated concentrations, as outlined in BSC Appendix A (2005ak). These particular experiments do not significantly influence the K_d distributions. The NRC staff notes that the experimental approaches are reasonable. Those experiments involving initial radionuclide concentrations higher than their solubility limit are not considered in the evaluation. DOE also addressed data uncertainty by obtaining sorption data that assessed K_d variability as a function of time, radioelement concentration, atmospheric composition, water composition, particle size, and temperature. Although most of the data were gathered from batch sorption experiments, DOE also performed a limited number of confirmatory column tests on selected radionuclides that DOE had identified as important contributors to mean annual dose in previous performance assessments, as outlined in SAR Section 2.3.9.3.2.2 and SNL Table 4-1 (2007ba).

The NRC staff review of DOE's sorption data and models indicates most of the K_d uncertainty distributions specified by DOE underpredicted the effectiveness of sorption compared to the experimental distributions. In some cases, DOE reduced the upper bounds of the K_d distributions (specifically, those of cesium, plutonium, and radium) relative to the range indicated by available data to account for the possible effects of slow sorption kinetics for these elements, as shown in BSC Appendix A (2005ak). By using low ranges of K_d values for sorption, DOE's transport model underpredicts retardation, resulting in relatively faster modeled travel times for radionuclides through the saturated zone. Following guidance in the YMRP, the NRC staff considered whether the use of conservative values to address uncertainty and variability of K_ds led to conservative estimates of risk and did not cause unintended results (i.e., did not introduce undue risk dilution or underestimate overall risk in the system performance). This aspect of biasing K_d distributions is described and evaluated later in this section when radioactive decay of parents and ingrowth of daughters is considered.

The empirical K_d modeling approach implemented by DOE is simplistic but well established (e.g., Freeze and Cherry, 1979aa; Till and Meyer, 1983aa) and has been broadly used to describe radionuclide transport (e.g., Sheppard and Thibault, 1990aa). One assumption inherent in the K_d modeling approach for sorption and the convolution integral approach used to calculate mass released from the saturated zone is that sorption and system behavior are linear (BSC, 2005ak). The ranges of K_d distributions have been adjusted to focus on the expected linear range of sorption, but the data DOE provided do not clearly indicate whether the adjustments are sufficient for each radioelement, as described in BSC Appendix A (2005ak). Another potential model uncertainty associated with the K_d approach is that individual K_d values are lumped parameters that do not explicitly take into account spatial and temporal variability or the role of specific surface-related processes that may affect radionuclide sorption. DOE addressed model uncertainty in TSPA calculations by sampling K_d values stochastically from uncertainty distributions in which the distribution ranges were developed from expected system conditions and conducted additional analyses to evaluate the effects of model scale and heterogeneity, as outlined in SNL Appendices C and D (2007ba). Rather than sample the K_d distribution independently for each radionuclide, DOE developed a correlation matrix for the 11 sorbing radioelements on the basis of their ranked sensitivities to six variables (pH, Eh, water chemistry, rock composition, rock surface area, and radionuclide concentration). DOE used this approach to approximate similarities in sorption behavior among radioelements and to ensure that transport behaviors were represented consistently within a single realization of the model, as outlined in SNL Appendix A (2007ba). The method of correlation did not appear to consider the potential for negative correlations (e.g., the sorption of two radionuclides is affected by pH, but in opposite directions). In addressing model uncertainty, DOE neglected sorption (i.e., K_d = 0) in fractures (fast flow paths), except for fault zones, and implemented K_d uncertainty distributions for matrix sorption. In most cases, this underpredicted the effectiveness of sorption compared to measured distributions so that, as modeled, radionuclide transport was less impeded by sorption than would otherwise be expected.

DOE developed information from natural analogs and field testing to provide qualitative comparisons for sorption model confidence building at the field scale (SAR Section 2.3.9.3.4.1.3). DOE did not apply the saturated zone transport abstraction to sorption modeling for the analog sites, nor did DOE use results from natural analog studies to inform the K_d distributions. However, DOE used general observations of sorption-related transport behavior to support the conceptual models (e.g., SAR Section 2.3.9.3.4.1.4). DOE also used observations from field tests at Busted Butte south of Yucca Mountain, from the C-Wells location, and from two alluvium tracer tests to provide qualitative and limited quantitative evaluations of sorption in the radionuclide transport model abstraction.

The NRC staff recognizes that, in general, underpredicting K_ds tends to allow for relatively faster transport times in the saturated zone, and thus earlier breakthrough of transported contaminants, and potentially greater modeled releases of radionuclides to the accessible environment. This approach can be conservative relative to performance, especially for the 10,000-year period, when DOE models predict that most radionuclides do not reach the accessible environment in that timeframe. This approach may not be conservative for a 1-million-year period, because during the longer period for radionuclides to reach the accessible environment, the difference in breakthrough time is less significant. For this longer period, one effect of relatively larger K_d values is that sorption can result in accumulation of some radionuclides in alluvium along the transport path and in the vicinity of the receptor location (albeit later in time). The use of low-biased K_d values increases the uncertainty in predicting the timing and magnitude of the peak dose over 1 million years.

The NRC staff performed independent transport simulations to confirm DOE's assumption that its choice of K_d distributions would not significantly affect its TSPA results for radionuclide transport. The NRC staff's transport simulations (Bradbury, 2010aa) involving radioactive decay and ingrowth suggest that biasing K_ds of certain radionuclides to lower values may lead to an underestimation of total radionuclide concentrations, when considering the sum of the parents and their ingrown daughters. In its response to requests for additional information, DOE (2009de) acknowledged that K_ds affect radionuclide concentrations. The analysis in DOE (2009de), however, used the biased distributions of K_ds from the TSPA, as described in BSC Appendix A (2005ak). The NRC staff used its independent transport simulations to consider the sensitivity of TSPA results to the K_d distributions. Using the information from DOE's sorption/desorption experiments, as outlined in SNL Appendix J (2007ba), the NRC staff reviewed the potential effects of higher K_ds on concentrations of parent-daughter radionuclides during saturated zone transport. These effects are most important only over long time periods, in transport calculations for one million-years. The NRC staff sensitivity analyses using pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) modeling program, the same computer code DOE used to model performance of the site, indicate that over such times total concentrations of the radionuclides of some decay series increase as a function of K_d of the parent. For example, the distribution used by DOE's TSPA for the K_d of uranium has a mean of ~5 mL/g. Increasing the K_d of uranium to 200 mL/g, similar to values of the effective K_d in DOE experiments, as outlined in BSC Appendix J (2005ak), increases the total flux of parent plus daughters up to a factor of 12, compared to using the lower mean K_d for uranium.

The impact of this excess activity has only a limited effect on performance. In this example, the calculated dose from the increase of these radionuclides is on the order of 0.1 mSv [10 mrem], as estimated by scaling the calculated dose by the relative increase in fluxes of daughter radionuclides. To arrive at this estimate, NRC staff followed a method for determining the effect of increased flux on dose similar to that used by DOE (2009de). This method begins with the DOE TSPA results for the contributions to dose by individual radionuclides, shown in SAR Figure 2.4-20b. As shown in this figure, DOE calculated the total mean annual dose, at 1 million years after repository closure, to be approximately 0.02 mSv [2 mrem]. The principal contributing radionuclides to this dose are Pu-242 {~0.006 mSv/yr [0.6 mrem/yr]}, Np-237 {~0.004 mSv/yr [0.4 mrem/yr]}, and I-129 {~0.002 mSv/yr [0.2 mrem/yr]}. Daughter radionuclides of the uranium decay series account for most of the remaining calculated annual dose. Increasing this remaining dose component by the increased flux of uranium daughter products (in this example, by a factor of ~12) gives an estimated total mean annual dose on the order of 0.1 mSv [10 mrem]. Note that this dose estimate is based on a single value for uranium sorption and is not weighted by the probability of occurrence.

Similar NRC analyses to determine the impact for possibly higher sorption of Np-237 suggest that an increased K_d could result in relatively lesser increases in total concentrations of parent plus daughters. Considering the relative contribution to total flux of these radionuclides, the NRC staff notes that biasing K_ds of the parents has only a limited impact on the total calculated total annual flux.

The NRC staff notes that in DOE's analysis, the pipe elements representing the alluvium were between 6.5 and 8.5 km [4.04 and 5.28 mi] long. For each realization, a single K_d for each radionuclide is assigned to the full extent of the alluvium. The NRC staff analyzed the effects of pipe length, and spatial heterogeneity of sorption on radionuclide release of parents and daughters to determine the sensitivity of DOE's approach to these parameters. The NRC staff analyses suggest that spatial heterogeneity does affect radionuclide concentrations for the post-10,000-year period, but the magnitude of this impact is not significant to performance, unless specific local geochemical conditions exist where sorption behavior is well outside the nominal ranges. The NRC staff evaluation of potential effects of local differences in sorption properties on specific decay chain radionuclides, and the likelihood of extreme values, is in TER Section 2.2.1.3.9.2.5.

In summary, DOE used experimental techniques with site-specific materials, alternative computer models, field tests, and natural analogs to provide a technical basis to support the TSPA model abstraction of radionuclide sorption. DOE used site-relevant data to address the anticipated effects of pH, Eh, major ion water chemistry, rock composition, rock surface area, and radionuclide concentration on radionuclide sorption. DOE acknowledged the limitations of the K_d approach and used stochastically sampled K_d probability distributions and simplifying assumptions about the effectiveness of sorption to address model and data uncertainty. DOE considered appropriate geochemical and physical conditions in developing the K_d distributions. DOE described the method used to assess the sensitivity of radioelement sorption behavior to variability in geochemical and physical conditions. DOE used these rankings to correlate sorption characteristics among the radioelements, ensuring consistency among the sorption parameters for each model realization. However, DOE described that the batch sorption tests provide results that underestimate the sorption of radionuclides expected in the field, compared to results from their desorption experiments, as described in BSC Appendix J (2005ak). DOE used distributions of K_d values in TSPA that overlap the batch sorption experiments results, but biased the distributions to lower values. As discussed previously, this method of biasing K_d distributions from sorption batch tests can underestimate concentrations of some daughters for disequilibrium in certain decay chains (DOE, 2009de). This leads to greater uncertainty in the calculated concentrations of the daughter radionuclides in groundwater. As previously discussed, the NRC staff analyses confirm DOE's assumption that the impact of these potential greater uncertainties is not significant for performance. The NRC staff, therefore, notes that while the DOE treatment of sorption might not be conservative for the million-year performance period, it is reasonable.

2.2.1.3.9.3.2.3 Matrix Diffusion

Diffusion is a physical process in which dissolved radionuclides move from a region of high concentration to a region of low concentration without advective flow. DOE described matrix diffusion as a fracture–matrix interaction that uses diffusion to transfer radionuclides between the water in fractures and the water in the rock matrix. DOE identified matrix diffusion as a moderately important transport mechanism in the saturated zone transport abstraction, especially for strongly sorbing radionuclides, because it is the main process by which radionuclides can move from a fracture-dominated flow path into the matrix. The modeled effectiveness of matrix diffusion depends on (i) the matrix diffusion rate (i.e., the rate that a radionuclide can diffuse from the water in a fracture into water in the pore spaces of the rock matrix) and (ii) the area of the fracture–matrix interface across which diffusion occurs, as outlined in SNL Section 6.1.2.4 (2007bj). In turn, the matrix diffusion rate depends on the concentration gradient of the radionuclide between fracture and matrix, and the value of the effective matrix diffusion coefficient, which is a measure of how readily a particular radioelement diffuses through a tortuous pathway of interconnected pores in the rock matrix. DOE estimated tortuosities from empirical data for representative Yucca Mountain tuff samples and developed standard normal cumulative probability distributions for effective matrix diffusion coefficients that were sampled stochastically in TSPA for each radioelement with respect to the individual model units (SAR Section 2.3.9.3.2; Reimus, et al., 2007aa).

In contrast to fractures in the unsaturated zone, where not all connected fractures in unsaturated rocks are water-bearing at the same time, all fractures in the saturated zone are, by definition, water bearing. However, as described in TER Section 2.2.1.3.9.3.2.1, not all fractures of the saturated zone aquifer participate in the flowing system. The flowing interval spacing reduces the number of fractures contributing to flow. This is a site characteristic measured in the field. Drill core logging yields spacing between fractures. Downhole spinner tests at packed locations yield spacing of flowing intervals. In the Yucca Mountain vicinity, the flowing interval spacing is greater than that of the fractures.

The NRC staff's evaluation of the information DOE provided about matrix diffusion in SAR Section 2.3.9.3.2.1 and relevant references considers staff's independent understanding of matrix diffusion models, the hydrogeologic characteristics of the saturated zone at the Yucca Mountain site, and field and laboratory studies of fracture–matrix interactions in saturated fractured rocks at Yucca Mountain and elsewhere. The NRC staff notes that DOE adopted an established theoretical approach to estimate radionuclide-specific effective matrix diffusion coefficients and, in developing the parameter values for matrix diffusion coefficients, DOE appropriately (i) synthesized Yucca Mountain geological and hydrological data, (ii) adapted the estimated values for saturated conditions, and (iii) accounted for uncertainty and natural variability in diffusion characteristics of different rock units. The NRC staff notes that DOE included sufficient data and addressed data uncertainty in developing effective matrix diffusion coefficients for the saturated zone transport abstraction.

DOE accounted for saturated zone transport-related model uncertainties by sampling values for the flowing interval spacing, the fracture porosity, and the effective diffusion coefficient in volcanic units. DOE's field experiments at the C-Wells Complex included cross hole tests where tracers with distinct diffusion coefficients were simultaneously injected into one well and the breakthrough curves of the different tracers were measured in a pumped well 30 m [32.8 yd] from the first. The differences in the breakthrough curves for the various tracers were used to demonstrate matrix diffusion was affecting tracer migration. The NRC staff notes that these field experiments provide confirmation that matrix diffusion is a process of attenuation in the volcanic rocks of the saturated zone because the experiments were designed to identify the process, if present.

The NRC staff compared DOE's large-scale tracer tests with published results from other saturated zone field experiments, modeling studies, and natural analogs at Yucca Mountain and elsewhere. In the context of data and model uncertainty about matrix diffusion as a saturated zone transport process, the NRC staff notes that DOE's performance assessment results illustrate that the retardation effect of matrix diffusion in the performance assessment is limited primarily to moderately and strongly sorbing radionuclides. The TSPA results with and without matrix diffusion in the saturated zone are comparable. Consequently, consistent with a risk-informed, performance-based technical evaluation, the NRC staff notes that DOE has provided a technical basis for DOE's representation of matrix diffusion in the saturated zone transport abstraction, consistent with YMRP guidance.

2.2.1.3.9.3.2.4 Colloid-Associated Transport

As described in TER Section 2.2.1.3.7.3.2.4, colloids are minute solid particles of any origin or composition that become suspended in a liquid. Because colloids are mobile in water, a radionuclide that is attached to a colloid (e.g., by sorption to the colloid surface) will be transported by the processes that move the colloid instead of by processes that would otherwise delay transport of the radionuclide as a dissolved species. Moreover, radionuclides attached to colloids tend to be transported preferentially in fast flow zones such as fractures or large pore throats because the size of colloids, compared to dissolved species, inhibits the transfer of colloids into fine-grained matrix, as described in SNL Section 6.8.2 (2008an).

DOE modeled colloidal transport in the saturated zone consistent with modeling used elsewhere in TSPA, with two types of radionuclide attachment: reversible and irreversible (BSC, 2005ak). Colloids with irreversibly attached radionuclides were modeled as separate transported entities, with a retardation factor applied specifically to the fractured volcanic tuff and alluvial aquifers to simulate the effects of nonpermanent filtration (BSC, 2005ak); DOE assumed that the size of irreversible colloids could exceed that of the pores of the volcanic matrix. Consequently, matrix diffusion of irreversible colloids in the saturated zone was neglected in the transport abstraction (BSC, 2005ak). Plutonium and americium were modeled as associated with both irreversible colloids and reversible colloids and as dissolved species in the saturated zone transport model, consistent with their transport mode in the unsaturated zone model. Reversible colloidal transport was modeled using the K_c factor, which represented equilibrium sorption of aqueous radionuclides onto natural system colloids (BSC, 2005ak). Radionuclides associated with reversible colloid transport comprised 4 of the 12 radionuclide groups modeled in the saturated zone flow and transport abstraction (BSC, 2005ak). These four groups included (i) plutonium, (ii) cesium, (iii) tin, and (iv) americium, protactinium, and thorium (BSC, 2005ak). Application of the K_c factor and inclusion of reversible sorption to colloids lowered the effective diffusion coefficient and the sorption coefficient, K_d, for the radionuclides (BSC, 2005ak), enhancing advective transport. The NRC staff notes that this approach is reasonable because cross hole field tests using microspheres show decreased retardation relative to reactive constituents. The NRC staff notes that the radioelements chosen for colloidal-facilitated transport are consistent with YMRP guidance because these radioelements are most strongly sorbed and, given low colloid concentrations, those radioelements that are the most strongly sorbed to the colloids contribute the most to release.

DOE included colloid-associated transport in both the three-dimensional saturated zone flow and transport abstraction and the one-dimensional saturated zone transport model (BSC, 2005ak). In the TSPA model abstraction, radionuclide mass exiting the unsaturated zone was partitioned into solution and onto colloids for transport in the saturated zone, as outlined in SNL Section 6.1.4.9 (2008ag). Irreversible colloids leaving the unsaturated zone were passed to the saturated zone transport abstraction as a single, irreversible colloid flux. For saturated zone transport calculations, the irreversible colloid flux was divided into a "slow" irreversible colloid fraction that is subject to modeled retardation processes during transport and a much smaller "fast" irreversible colloid fraction that was assumed to travel unretarded throughout the saturated zone. Colloid-associated transport of radionuclides is affected by filtration, the rate of desorption from the colloid, and the colloid concentrations in the groundwater (SAR Section 2.3.9.3). Each of these factors was included in the saturated zone colloid-associated transport model.

The DOE colloid-associated transport model treats radioactive decay in irreversible colloids by assuming that if a decay product was also one of the two radioelements associated with an irreversible colloid in the model (i.e., isotopes of plutonium and americium), then the decay product remained irreversibly associated with the colloid (SAR Section 2.3.9.3.2.3). Otherwise, the decay product enters the aqueous phase as a dissolved species (SAR Section 2.3.9.3.2.3). In the model abstraction there was no permanent filtration of irreversible colloids due to size exclusion in the tuff matrix, at the transition from tuff to alluvium, or in the alluvium, so no colloid size parameter was required in the saturated zone transport models (SAR Section 2.3.9.3.2.3). The nonpermanent filtration of irreversible colloids was implicitly included as part of the basis and development of the irreversible colloid retardation factor for both the tuff and the alluvium (SAR Section 2.3.9.3.2.3; BSC, 2005ak, 2004bc). These approaches to simulating colloid transport used consistent assumptions for parent and daughter. The DOE model that does not consider permanent filtration of irreversible colloids allows for larger releases of colloid-associated radionuclides at the expense of accumulated masses of colloid-associated radionuclides in the saturated zone (see TER Section 2.2.1.3.9.3.2.5 concerning ingrowth).

DOE's conceptual model assumed that reversible colloids could be represented by particles with the composition and characteristics of the clay mineral montmorillonite. Although naturally occurring colloids in Yucca Mountain groundwaters consist of montmorillonite, zeolite, and silica, the NRC staff notes the use of montmorillonite is reasonable, as the specific mineral is less significant than the assigned sorption coefficients. DOE addressed data uncertainty for sorption onto reversible colloids by selecting a reasonable range of montmorillonite sorption coefficients, which captures that of other colloid minerals.

DOE developed the uncertainty distributions for the concentration of groundwater colloids from data collected in saturated zone field studies from the Yucca Mountain region and from groundwater analyses elsewhere (BSC, 2005ak; SNL, 2007aw,bi). The colloid concentrations represented in the model covered a broad range of values that account for higher colloid concentrations measured in some groundwaters, with these higher concentrations given a low probability of occurrence. Colloids were assumed to be stable for all water chemistry conditions in the saturated zone. The NRC staff notes that the assumptions for colloid concentrations and stability in the saturated zone are consistent with groundwater analyses observations for the Yucca Mountain region.

The only radioelements irreversibly associated with colloids in DOE's model are plutonium and americium. After the general corrosion failure of waste containers in TSPA simulations, up to 30 percent of the Pu-242 flux transported to the accessible environment is by irreversible colloids (e.g., SAR Section 2.4.2.2.3.2.2 and Figure 2.4-108). On the basis of data from field experiments that some colloids travel with little or no retardation (Kersting, et al., 1999aa; SNL, 2007aw), DOE designated a small fraction (less than 0.2 percent) of the irreversible colloid flux as a completely unretarded "fast fraction."

In reviewing DOE's technical basis for colloid-associated transport in the saturated zone, the NRC staff evaluated information DOE provided in SAR Section 2.3.9 and references therein (BSC, 2004bc; SNL, 2008ag,an, 2007bi). The NRC staff also considered additional information in DOE Enclosures 9–14 (2009am) and the NRC staff's independent experience with colloid-associated transport processes and models in heterogeneous natural systems, such as the saturated zone at Yucca Mountain.

On the basis of the information DOE provided in SAR Section 2.3.9, the NRC staff notes that DOE provided a technical basis for the representation of radionuclide transport by reversible colloids. DOE accounted for system variability in developing parameter values, where feasible, from site-specific data from saturated zone field tests in the Yucca Mountain area and sampling colloid-associated parameter values from large uncertainty distributions. The NRC staff notes that DOE addressed model uncertainty because the results are consistent with the NRC staff's understanding of colloid-associated transport processes and the uncertainties involved in characterizing colloidal transport processes in natural systems. DOE's approach used reasonable distributions of parameter values, simple model abstractions supported by field and lab tests, and analyses of natural analogs and underground nuclear tests.

With respect to DOE's representation of radionuclide transport by irreversible colloids in the model, the NRC staff notes that DOE's model is reasonable because it includes only processes that have been demonstrated to be present in field tests and lab experiments. DOE's treatment of colloid-associated transport is consistent with its model for partitioning of the radioelements among the three transport entities (dissolved species, reversibly associated with colloid, and irreversibly associated with colloid), which is evaluated in TER Section 2.2.1.3.4.

In summary, DOE has developed a conceptual and mathematical basis for colloid-associated transport processes in the saturated zone (e.g., retardation of colloids by attachment processes in fractured volcanic tuff and alluvium, reversible sorption of radionuclides onto colloids, colloid exclusion processes, and unretarded colloidal transport) that is consistent with existing models for contaminant transport in fractured rocks and porous media in the literature (e.g., Sudicky and Frind, 1982aa). DOE's modeling approach compensated for the high uncertainty in empirical observations for saturated zone colloidal transport in field studies or natural analogs by using reasonable probability distributions for most colloid-related parameters.

2.2.1.3.9.3.2.5 Radionuclide Decay and Ingrowth

Radioactive decay is a general term for the processes by which unstable radionuclides spontaneously disintegrate to form a different nuclide that may or may not be radioactive. Loss of radionuclides over time due to radioactive decay and, where applicable, the potential ingrowth (increase) of radionuclide daughters were included in the DOE saturated zone transport abstraction. Several heavy radionuclides are parents to decay chains of multiple radioactive daughters. In the absence of chemical fractionation, the radionuclides in the chain reach secular equilibrium, where parents and daughters have equal activity. Disequilibrium of naturally occurring decay chains is observed in many groundwater systems, due to geochemical processes in the aquifers (e.g., Faure, 1986aa).

Unlike its unsaturated zone transport abstraction, DOE did not directly calculate radioactive decay and ingrowth processes in its three-dimensional site-scale saturated zone transport model. Instead, decay was included in the convolution integral model during the calculation of mass breakthrough. Moreover, the three-dimensional saturated zone transport model as formulated did not explicitly include the ingrowth of decay progeny. The mass releases of the radionuclides C-14, Cs-135, Cs-137, I-129, Sr-90, Tc-99, Am-243, Pu-239, Am-241, Pu-240, Pu-242, Pu-238, Cl-36, Se-79, Sn-126, Np-237, U-234, U-232, U-236, and U-238, were determined from the results of the three-dimensional transport model. DOE used two mechanisms to account for daughter ingrowth.

First, at the transition from the unsaturated zone to the saturated zone, the masses of first-generation daughters, Np-237, U-234, U-236, U-238, and Pu-239, were boosted by the amount their parents were expected to decay during the remainder of the simulated performance time period, as outlined in SNL Section 6.3.10.3 (2008ag). The parent and boosted daughter pairs are Am-241/Np-237, U-238/U-234, Pu-238/U-234, Pu-240/U-236, Pu-242/U-238, and Am-243/Pu-239.

The NRC staff notes that this method can underestimate or overestimate the amount of daughter in the saturated zone, depending on the transport characteristics of the parent and daughter. At early times in the saturated zone transport model, the daughter concentrations could be overestimated, because DOE's inventory-boosting methodology assumes all of the ingrowth occurs when the parent radionuclide enters the saturated zone. At later times, the daughter concentrations may be underestimated, because a quickly transported (low K_d) daughter may exit the model system before the parent would have actually generated the daughter via decay. The NRC staff notes that one-dimensional transport simulations using the pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) computer code are reasonable because they adequately represent the important processes controlling radionuclide transport and concentrations. The NRC staff compared one-dimensional transport simulations of the parents and the first-generation daughters with radioactive decay and ingrowth to one-dimensional transport simulations with decay and boosted daughters instead of ingrowth (Bradbury, 2010ab). The NRC staff notes that the inventory-boosting method is reasonable because the results from the two types of simulation were comparable at later times when steady state was established. Prior to steady state, the inventory-boosting method overestimated the mass of the daughter in the saturated zone.

DOE used its one-dimensional saturated zone transport model to calculate the ingrowth of second-generation daughter radionuclides in selected decay chains (SAR Section 2.3.9.3.4.2; BSC, 2005ak). The mass of secondary daughters the one-dimensional model transported through the saturated zone was added to the mass of radionuclides the three-dimensional model transported to determine the total mass of radionuclides transported through the saturated zone. The mass releases of the radionuclides U-235, U-233, Th-230, Pa-231, Th-229, Th-232, and Ra-226 were determined from the results of the one-dimensional transport model. The DOE model assumed that Ac-227, Ra-228, Th-228, and Pb-210 were in secular equilibrium with their dissolved parents. DOE's assumptions of secular equilibrium in groundwater are unsupported for parents and daughters that have substantially different sorption properties.

This effect is further analyzed in DOE (2009de). DOE's analysis showed that the K_d of the parent divided by the K_d of the daughter times the activity of the parent in groundwater can be used to determine the activity of the daughter in groundwater. In DOE (2009de), DOE calculated that the activity of Ra-228 in groundwater is on average 14 times greater than that of its parent Th-232, and the activity of Ac-227 in groundwater is on average 6.8 times greater than that of its parent Pa-231. The activity of Pb-210 in groundwater is approximately the same as its distant parent, Ra-226, due to their comparable K_d distributions. An extreme example showing the effect of K_ds on activities of parent and daughter is the Ra-226/Rn-222 pair. Radon, an inert gas, is assumed not to sorb, whereas radium strongly sorbs, as indicated by its uniform K_d distribution from 100 to 1,000 mL/g. DOE calculated the activity of Rn-222 to be, on average, 1,400 times that of Ra-226 in groundwater. DOE (2009de) showed that these corrections to activity of these daughters were not enough to significantly increase dose {an increase from 0.02 to 0.024 mSv [2.0 to 2.4 mrem] for the maximum total mean annual dose}. In addition to the pairs DOE (2009de) considered, other parent-daughter pairs could have significantly different activities than those assumed in the TSPA model, depending on the differences in effective K_ds. For example, the NRC staff notes that, given reasonable K_ds for the two elements, the mean activity ratio of thorium and radium would be on the order of 14 for all the thorium/radium pairs, not just Th-232/Ra-228 (other pairs include Th-230/Ra-226, Th-228/Ra-224, Th-227/Ra-223, and Th-229/Ra-225). The NRC staff notes that the GoldSim (GoldSim Technology Group, 2006aa) pipe element correctly simulates sorption and transport of radioactive species. Th-230 and Ra-226 are transported in the saturated zone abstraction using the pipe element, so their activity ratio should correctly reflect the sorption enhancement factor. The other thorium/radium pairs are not explicitly modeled, so the same sorption enhancement factor corrections to the activity ratio are appropriate. The NRC staff notes that some daughter activities are less than those of the parent and some are more, depending on the K_ds of the parent and daughter.

The groundwater exposure scenario in the biosphere model assumes that short-lived daughters (half-life less than 180 days) are in secular equilibrium with their parent primary radionuclide. The biosphere model calculates biosphere dose conversion factors when the receptor uses water containing a nominal activity of the primary radionuclide. That water also is assumed to contain the same activity of each of the short-lived daughters. However, on the basis of DOE (2009de), when directed to the short-lived daughters, the same enhancement factor should apply. For example, given the difference in K_ds for thorium and radium, a receptor, on average, would be exposed to 14 times more Ra-225 activity than Th-229 (the activities would be the same under the assumption of secular equilibrium). Evaluations of biosphere pathways are described in TER Section 2.2.1.3.14.

An important aspect of the contribution of short-lived daughters to dose is that, in general, the calculated dose for a given activity decreases down the decay chain. Consequently, increasing activities of daughters when considering the sorption enhancement factor are in many cases compensated for by the decreased dose coefficient. Evaluations of biosphere dose coefficients are described in TER Section 2.2.1.3.14.

To confirm DOE's assumption that deviations from secular equilibrium would not significantly affect TSPA results, the NRC staff has performed analyses (Bradbury, 2010aa) using the pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) code to simulate transport of decay chain radionuclides including those short-lived radionuclides not included in DOE's model. The NRC staff notes that although the concentrations and activities of some radionuclides may be greater than described in the SAR, the resultant calculated dose for all radionuclides is low.

A source of uncertainty is the potential contribution to dose from Po-210, a daughter of Pb-210 in the U-238 decay chain. There are some observations to indicate that Po-210 may not be removed by sorption from natural waters as readily as its immediate parent, Pb-210 (Serne, 2007aa; Hameed, et al., 1997aa). At two locations in the United States, Po-210 has been measured in groundwater at concentrations greater than would be expected based on measured Pb-210 concentrations. These locations are an alluvial aquifer in Churchill County, Nevada (Seiler, 2011aa; Seiler, et al., 2009aa) and shallow aquifers of west central Florida (Harada, et al., 1989aa). One explanation of these Po-210 excesses is that the specific geochemical conditions present in these locations leads to differential sorption of parent and daughter radionuclides. For this decay chain, NRC staff analyses show that the difference in sorption of the much longer-lived parents at the head of the decay chain (U-238, U-234) relative to Po-210 is more significant than the relative sorption of the immediate parent (Pb-210) (Bradbury, 2010aa,ab).

For further confirmation of DOE's assumption that deviations from secular equilibrium would not significantly affect performance, the NRC staff reviewed scientific literature to understand the sorption behavior of Po-210 in natural systems under different conditions. Bradbury and Bayens (2003aa) used selenium as an analog for polonium to estimate a median K_d of 180 mL/g under reducing conditions (low oxygen) and neutral pH (~7). Experiments by Baston, et al. (1999aa) also conducted under reducing conditions, but at higher pH (~9.4), produced polonium K_ds on the order of 1,500 to 36,500 mL/g. Hameed, et al. (1997aa) derived a polonium K_d of 43,000 mL/g for sediment in an organic-rich pond. On the basis of a review of literature, Serne (2007aa) suggested a polonium K_d range of 150 to 1,100 mL/g for surface soils near the Hanford site. There is a relative lack of recent data for Po-210 sorption under oxidizing conditions, but the low concentrations of polonium typically measured in oxidizing groundwaters are consistent with high K_ds. There are indications that in some settings, polonium in groundwater is associated with colloids or other filterable particles (Serne, 2007aa; Vaaramaa, et al., 2003aa).

Although the behavior of polonium in groundwater is not fully understood, there are several geochemical parameters that appear common to locations where significant excess Po-210.has been recognized. On the basis of the NRC staff review, conditions associated with excess polonium in groundwater include microbial-mediated sulfur cycling (Nevada and Florida), very low oxygen concentrations (Nevada and Florida), alkaline (pH 9 or above; Nevada) or acidic (pH below 6; Florida) conditions, moderate to high organic carbon content (Nevada and Florida), detectable sulfide concentrations (Nevada and Florida), and iron or manganese colloids (Nevada and Florida) (Harada, et al., 1989aa; Burnett, et al., 1991aa; Upchurch, et al., 1991aa; Seiler, et al., 2009aa; Seiler, 2011aa). In the Yucca Mountain saturated zone system, groundwater near the boundary with the accessible environment is characterized by oxidizing waters of neutral pH (6.8 to 8.5 with a median of 7.85), with low organic content, no measurable sulfide, and little or no iron oxide or manganese oxide colloids. Conditions that would appear to favor excess Po-210 have not been observed along the potential flow path in the saturated zone alluvium south of the controlled boundary. Consequently, the NRC staff notes that excess Po-210, beyond the secular equilibrium values assumed by DOE, is unlikely to occur in the saturated zone such that it significantly impacts repository performance. To confirm DOE's approach regarding the likelihood of excess Po-210 occurring in the saturated zone, DOE should include in its performance confirmation program the following:

- Data that support the absence of the set of geochemical conditions described previously as associated with excess polonium in other locations, by analysis of existing water wells within Fortymile Wash and the adjacent alluvial basin, considering potential flow paths of radionuclide transport into the accessible environment

- Continued monitoring of those same wells for the set of geochemical conditions associated with excess polonium in other locations

- Experimental determination of polonium sorption on representative alluvial material or proxies in order to constrain appropriate polonium K_d values, under a range of relevant geochemical conditions

Given current understanding of the behavior of decay chain radionuclides in the saturated zone transport path at Yucca Mountain, the NRC staff notes that the DOE approach to radionuclide decay and ingrowth is reasonable because the uncertainties in radionuclide concentrations from potential accumulation of decay-chain parent radionuclides are not significant for performance of the saturated zone. DOE should confirm its approach for decay chain radionuclide behavior by providing through its performance confirmation program information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone.

Radioactive decay and ingrowth processes were modeled for dissolved, reversible colloids and irreversible colloid radionuclide species all of the types included in the saturated zone transport abstraction. The NRC staff notes that the inventory-boosting methodology accounts for decay from all parent sources in the saturated zone transport model. The NRC staff notes that DOE's treatment of decay chains in irreversible colloids is consistent with DOE's model assumptions about which radionuclides are associated with reversible and irreversible colloids.

2.2.1.3.9.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for radionuclide transport in the saturated zone is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA). DOE should confirm its approach for decay chain radionuclide behavior by providing through its performance confirmation program information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone, as identified in TER Section 2.2.1.3.9.3.

2.2.1.3.9.5 References

Baston, G.M.N., J.A. Berry, M. Brownsword, T.G. Heath, D.J. Ilett, R. McCrohon, C.J. Tweed, and M. Yui. 1999aa. "The Sorption of Polonium, Actinium, and Protactinium Onto Geological Materials." Materials Research Society Symposium Proceedings 556. Warrendale, Pennsylvania: Materials Research Society. pp. 1,107–1,122.

Bradbury, J.B. 2010aa. "Secular Equilibrium Assumption Analysis." Scientific Notebook. ML110820653. Washington, DC: U.S. Nuclear Regulatory Commission.

Bradbury, J.B. 2010ab. "Boosted Inventory Assumption Analysis." Scientific Notebook. ML110820655. Washington, DC: U.S. Nuclear Regulatory Commission.

Bradbury, M.H. and B. Bayens. 2003aa. "Far Field Sorption Data Bases for Performance Assessment of a High-Level Radioactive Waste Repository in an Undisturbed Opalinus Clay Host Rock." Bericht Nr. 03-08, Nagra NTB 02-19. Villigen, Switzerland: Paul Scherrer Institut.

Burnett, W.C., J.B. Cowars, W.-C. Tai, and P.H. Cable. 1991aa. “Investigations of Radon and Radon Daughters in Surficial Aquifers of Florida.” No. 05-032-093. Bartow, Florida: Florida Institute of Phosphate Research.

BSC. 2005ak. “Saturated Zone Flow and Transport Model Abstraction.” MDL–NBS–HS–000021. Rev. 03. AD 01, AD 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004bc. “Saturated Zone Colloid Transport.” ANL–NBS–HS–000031. Rev. 02. ACN 01, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Davis, J.A. and G.P. Curtis. 2003aa. NUREG/CR–6820, “Application of Surface Complexation Modeling To Describe Uranium (VI) Adsorption and Retardation at the Uranium Mill Tailings Site at Naturita, Colorado.: Washington, DC: NRC.

Davis, J.A. and D.B. Kent. 1990aa. “Surface Complexation Modeling in Aqueous Geochemistry.” Mineral-Water Interface Geochemistry. M.F. Hochella, Jr. and A.F. White, eds. *Mineralogy.* Vol. 23. pp. 177–260.

Davis, J.A., J.A. Coston, D.B. Kent, and C.C. Fuller. 1998aa. “Application of the Surface Complexation Concept to Complex Mineral Assemblages.” *Environmental Science & Technology*. Vol. 32. pp. 2,820–2,828.

DOE. 2009am. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.8), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.7, Set 1.” Letter (February 9) J.R. Williams to J.H. Sulima (NRC). ML090410352. Washington, DC: DOE, Office of Technical Management.

DOE. 2009de. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.9), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.9, Set 1.” Letter (November 16) J.R. Williams to J.H. Sulima (NRC). ML092820675. Washington, DC: DOE, Office of Technical Management.

DOE. 2009df. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.9), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.9, Set 1.” Letter (October 9) J.R. Williams to J.H. Sulima (NRC). ML093210213. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, “Yucca Mountain Repository License Application.” Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Faure, G. 1986aa. *Principles of Isotope Geology, 2nd Edition.* New York City, New York: John C. Wiley & Sons, Inc.

Freeze, R.A. and J.A. Cherry. 1979aa. *Groundwater.* Englewood Cliffs, New Jersey: Prentice-Hall, Inc.

GoldSim Technology Group. 2006aa. *GoldSim Contaminant Transport Module User’s Guide.* Version 4.0. Issaquah, Washington: GoldSim Technology Group.

Hameed, P.S., K. Shaheed, and S.S.N. Somasundaram. 1997aa. “A Study on Distribution of Natural Radionuclide Polonium-210 in a Pond Ecosystem.” *Journal of Biosciences*. Vol. 22, No. 5. pp. 627–634.

Harada, K. W.C. Burnett, J.B. Cowart, and P.A. LaRock. 1989aa. “Polonium in Florida Groundwater and Its Possible Relationship to the Sulfur Cycle and Bacteria.” *Geochimica et Cosmochimica Acta*. Vol. 53, No. 1. pp. 143–150.

Kersting, A.B., D.W. Efurd, D.L. Finnegan, D.J. Rokop, D.K. Smith, and J.L. Thompson. 1999aa. “Migration of Plutonium in Ground Water at the Nevada Test Site.” *Nature*. Vol. 397, No. 6714. pp. 56–59.

Langmuir, D. 1997aa. *Aqueous Environmental Geochemistry*. Upper Saddle River, New Jersey: Prentice-Hall, Inc.

NRC. 2009ab. “Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63.” Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. Washington, DC: NRC.

Reimus, P.W., T.J. Callahan, S.D. Ware, M.J. Haga, and D.A. Counce. 2007aa. “Matrix Diffusion Coefficients in Volcanic Rocks at the Nevada Test Site: Influence of Matrix Porosity, Matrix Permeability, and Fracture Coating Minerals.” *Journal of Contaminant Hydrology*. Vol. 93. pp. 85–95.

Seiler, R.L. 2011aa. “210Po in Nevada Groundwater and Its Relation to Gross Alpha Radioactivity.” Ground Water. Vol. 29, No. 2. pp. 160–171.

Seiler, R.L., L.L. Stillings, and N. Cutler. 2009aa. “Mobilization of Polonium-210 in Naturally-Contaminated Groundwater, Churchill County, Nevada.” Presentation at American Geophysical Union 2009 Fall Meeting, San Francisco, California, December 14–18, 2009. Washington, DC: American Geophysical Union.

Serne R.J. 2007aa. “K_d Values for Agricultural and Surface Soils for Use in Hanford Site Farm, Residential, and River Shorelines Scenarios.” PNNL–16351. Richland, Washington: Pacific Northwest National Laboratory.

Sheppard, M.I. and D.H. Thibault. 1990aa. “Default Soil Solid/Liquid Partition Coefficients, K_ds, Four Major Soil Types: A Compendium.” *Health Physics*. Vol. 59. pp. 471–482.

SNL. 2008ag. “Total System Performance Assessment Model/Analysis for the License Application.” MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008an. “Particle Tracking Model and Abstraction of Transport Processes.” MDL–NBS–HS–000020. Rev. 02. ADD 02. Las Vegas Nevada: Sandia National Laboratories.

SNL. 2007ah. “Dissolved Concentration Limits of Elements with Radioactive Isotopes.” ANL–WIS–MD–000010. Rev. 06. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007aw. "Saturated Zone *In-Situ* Testing." ANL–NBS–HS–000039. Rev. 02. ACN 01, ACN 02, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ba. "Site-Scale Saturated Zone Transport." MDL–NBS–HS–000010. Rev. 03. ACN 01, AD 001. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bi. "Waste Form and In-Drift Colloids-Associated Radionuclide Concentrations: Abstraction and Summary." MDL–EBS–PA–000004. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007bj. "Radionuclide Transport Models Under Ambient Conditions (U0060)." MDL–NBS–HS–000008. Rev. 02. CAN 002, AD 01. Las Vegas, Nevada: Sandia National Laboratories.

Sudicky, E.A. and E.O. Frind. 1982aa. "Contaminant Transport in Fractured Porous Media: Analytical Solutions for a System of Parallel Fractures." *Water Resources Research*. Vol. 18. pp. 1,634–1,642.

Till, J.E. and H.R. Meyer, eds. 1983aa. NUREG/CR–3332, "Radiological Assessment: A Textbook on Environmental Dose Analysis." Washington, DC: NRC.

Turner, D.R. and R.T. Pabalan. 1999aa. "Abstraction of Mechanistic Sorption Model Results for Performance Assessment Calculations at Yucca Mountain, Nevada." *Waste Management*. Vol. 19. pp. 375–388.

Turner, D.R., F.P. Bertetti, and R.T. Pabalan. 2002aa. "The Role of Radionuclide Sorption in High-Level Waste Performance Assessment: Approaches for the Abstraction of Detailed Models." *Geochemistry of Soil Radionuclides*. P.-C. Zhang and P.V. Brady, eds. Special Publication 59. Madison, Wisconsin: American Society of Agronomy. pp. 211–252.

Upchurch, S.B., C.R. Oural, D.W. Foss, and H.R. Brooker. 1991aa. "Radiochemistry of Uranium-Series Isotopes in Groundwater." No. 05-022-092. Bartow, Florida: Florida Institute of Phosphate Research.

Vaaramaa, K., J. Lehto, and H. Ervann. 2003aa. "Soluble and Particle-Bound $^{234, 238}U$, ^{226}Ra, and ^{210}Po in Ground Waters." *Radiochimica Acta*. Vol. 91. pp. 21–27.

CHAPTER 13

2.2.1.3.10 Igneous Disruption of Waste Packages

2.2.1.3.10.1 Introduction

This Technical Evaluation Report (TER) chapter evaluates the U.S. Department of Energy's (DOE) models for the potential consequences of disruptive igneous activity at Yucca Mountain if basaltic magma rising through the Earth's crust intersects and enters a repository drift or drifts [DOE's igneous intrusion modeling case, Safety Analysis Report (SAR) Section 2.3.11.3 (DOE, 2009av)] or enters a drift and later erupts to the surface through one or more conduits (DOE's volcanic eruption modeling case, SAR Section 2.3.11.4). The proposed Yucca Mountain repository site lies in a region that has experienced sporadic volcanic events in the past few million years, such that DOE previously determined the probability of future igneous activity at the site to exceed 1×10^{-8} per year (SAR Section 2.2.1.2.2; CRWMS M&O, 1996aa). DOE therefore included igneous activity as one of three scenario classes in its performance assessment. Because basalt is the only type of magma that has been erupted in the past 8 million years in the Yucca Mountain region, DOE's performance assessment considers only basaltic igneous activity. As discussed in TER Section 2.2.1.2.2, the probability of more silicic (explosive) igneous activity, of the type that produced extensive pyroclastic deposits in the area more than 10 million years ago, is well below 1 in 10,000 over 10,000 years; DOE screened this out as a potential disruptive event.

This chapter evaluates the subsurface igneous processes DOE described (i.e., intrusion of magma into repository drifts, damage to waste packages and other engineered barriers, and formation of conduits to the surface, which involves entrainment of waste into the conduit and toward the surface). DOE's models for volcanic ejection and dispersal of waste material into the surface environment are reviewed in TER Section 2.2.1.3.13. Together, TER Sections 2.2.1.3.10 and 2.2.1.3.13 evaluate DOE information and output that is used in the Total System Performance Assessment (TSPA) under the Igneous Scenario Class (see TER Sections 2.2.1.2.1, 2.2.1.2.2, and 2.2.1.4.1).

DOE examined the consequences of igneous disruption of the repository (Igneous Scenario Class) using the results of TSPA calculations through the two linked modeling cases, igneous intrusion and volcanic eruption (intrusion always precedes eruption). DOE's igneous intrusion modeling case provides TSPA parameter values for the number of waste packages failed (mass of waste) during an intrusive event, the temperature in the invaded drifts in the period after intrusion, and chemical changes to groundwater that may react with the basalt filling the drifts. The igneous disruption of waste packages abstraction integrates with other TSPA model components, such as the unsaturated zone radionuclide transport abstraction, and provides information about the flux of radionuclides released from the waste form into water entering the unsaturated zone after an intrusive event (TER Section 2.2.1.3.7). Exposure to radionuclides in groundwater extracted by pumping is one of the principal pathways for radiological exposure to the reasonably maximally exposed individual (RMEI).

In the DOE volcanic eruption modeling case, a key parameter affecting the overall dose calculation is the number of directly affected waste packages and thus the amount of waste entrained in a volcanic eruption. DOE's model of the airborne transport and redistribution of radionuclides into soil includes the amount of waste erupted into the atmosphere, the amount deposited on the ground, and the redistribution of the waste-contaminated volcanic ash. This airborne transport and redistribution model is evaluated in TER Section 2.2.1.3.13 and provides information for the Volcanic Ash Exposure Scenario described in DOE's Biosphere Model (SAR Section 2.3.10). DOE's estimate of the annual probability of igneous events intersecting the repository (1.7×10^{-8} per year; SAR Table 2.3.11-4) is reviewed in TER Section 2.2.1.2.2 and briefly discussed later in this chapter. For these abstractions and the TSPA, DOE calculates probability-weighted results for both an intrusive-only dose and a total dose (intrusive plus volcanic) to the RMEI, which are detailed in TER Section 2.2.1.4.1 and outlined in the Risk Perspectives subsection in TER Section 2.2.1.3.10.3.1.

Igneous disruption models evaluated in this chapter are the first in a sequence of models that track radionuclides released from the repository to the RMEI as a result of possible future igneous activity. Accordingly, the model abstractions evaluated in this TER chapter serve as input to those reviewed in other chapters, including those that examine the effects of potential igneous disruption of natural and engineered barriers in the subsurface repository (TER Section 2.2.1.3.2). DOE recognized that igneous events potentially have large consequences but a low likelihood (probability) of occurring in the future (SAR Section 2.3.11.1). Thus, DOE provided only a qualitative description of igneous effects on engineered system barrier capabilities in its demonstration of multiple barriers (SAR Section 2.1.1). Nevertheless, basaltic igneous activity represents a disruptive event that significantly degrades most of the capabilities of the engineered barrier system (SAR Section 2.1.2.2.5). From review of the DOE information in SAR Section 2.1 relevant to the effects of igneous events on repository barrier capabilities, the U.S. Nuclear Regulatory Commission (NRC) staff notes that this information is consistent with the representation of igneous events in the performance assessment. To represent igneous events in the performance assessment, DOE removes the barrier capabilities of the waste package and drip shield and degrades the waste form, consistent with information provided in SAR Section 2.3.11. DOE further concluded in SAR Section 2.1.1 that igneous events will have limited effects on the upper and lower natural barrier systems because the possible igneous intrusive rock bodies have very small dimensions compared with the large volume of rock through which groundwater is flowing and the zone of influence around the intrusions is limited (SAR Section 2.1.2.3.5). NRC staff notes this is also consistent with the information in SAR Section 2.3.11 and discusses this aspect of DOE's performance assessment in a subsection of TER Section 2.2.1.3.10.3.2.

2.2.1.3.10.2 Evaluation Criteria

NRC staff's review of model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for igneous disruption of waste packages, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The resulting DOE Total System Performance Assessment (TSPA) is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of features, events, and processes is given in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following permanent closure. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The model abstraction of igneous disruption of waste packages involves igneous activity. 10 CFR 63.342(c)(1) provides requirements for assessing the effects of seismic and igneous activity on the repository performance, subject to the probability limits given in 10 CFR 63.342(a) and (b). Specific constraints on the seismic and igneous activity analyses are in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP), NUREG–1804, Section 2.2.1.3.10, Volcanic Disruption of Waste Packages, (NRC, 2003aa), as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's evaluation of DOE's abstraction of igneous disruption of waste packages are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

The NRC staff review used a risk-informed approach and the guidance provided by the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by NRC (2009ab), to the extent reasonable for aspects of igneous disruption of waste packages important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by NRC staff, are discussed in this chapter. The NRC staff's determination is based both on risk information provided by DOE, and on NRC staff knowledge gained through experience and independent analyses.

2.2.1.3.10.3 Technical Evaluation

DOE's analysis of potentially disruptive features, events, and processes (FEPs) considered ways that igneous activity could affect the proposed repository site. NRC staff evaluation of DOE's FEP screening is given in TER Section 2.2.1.2.1. DOE included the following FEPs and defined the igneous scenarios for the performance assessment (SAR Table 2.3.11-1): 1.2.04.03.0A, Igneous Intrusion into Repository; 1.2.04.04.0A, Igneous Intrusion Interacts with Engineered Barrier System Components; 1.2.04.06.0A, Eruptive Conduit to Surface Intersects Repository; and 1.2.04.04.0B, Chemical Effects of Magma and Magmatic Volatiles (SAR Table 2.2-5). This chapter evaluates repository performance as affected by these FEPs. Other included FEPs related to potential igneous activity are reviewed in TER Sections 2.2.1.3.2 and 2.2.1.3.13.

The NRC staff's review is based on information presented in SAR Section 2.3.11 and relevant analysis and model reports (AMRs), on material in other publicly available DOE and the NRC reports, and on relevant information published in peer-reviewed literature. DOE also described and evaluated background information used to assess the likelihood and style of future igneous activity in the Yucca Mountain region in SAR Volume 1, General Information, and Volume 2, Section 1.1.2. That material is reviewed in TER Section 2.1.1.1.3.6 as part of the NRC staff's evaluation of site characterization.

2.2.1.3.10.3.1 General Approach by DOE

Igneous activity can be solely intrusive (i.e., magma intruded into rocks below the Earth's surface) or extrusive (i.e., volcanic, in which, following intrusion, magma breaks through to the surface and an eruption ensues). The NRC staff notes that the terms "volcanic" and "intrusive" have sometimes been used interchangeably in the DOE SAR and supporting documents. To avoid confusion, the NRC staff will refer to igneous activity that occurs beneath the Earth's surface as "intrusive" and activity above surface as "extrusive" or "volcanic." All subsurface igneous processes beneath a possible future active volcano that could disrupt the repository are considered intrusive and are reviewed in this chapter, whereas the above-surface volcanic processes are discussed and reviewed in TER Section 2.2.1.3.13.

To evaluate the potential effect of future igneous activity on dose to the RMEI, DOE adopted a conceptual model in which rising basalt magma entering a repository drift (or drifts) could cause release of radionuclides via two pathways (SAR Section 2.3.11.1). During intrusive igneous events, magma rising toward the surface as a dike, or set of dikes, enters drifts but stays beneath the surface. DOE also considered the other type of small igneous intrusion, sills (relatively small subhorizontal igneous intrusions), but did not treat them separately in its analysis. One reason is that dikes must be present to feed magma into sills, and DOE showed that the consequences of intruding the repository by a sill would be similar to and more limited than by a dike. DOE also pointed to the relatively small size of sills in the Yucca Mountain region (Valentine and Krogh, 2006aa; Keating, et al., 2008aa) and thus did not include sills in its igneous disruption scenario for the repository (SAR Section 2.3.11.2.1.1). In the igneous intrusive scenario, DOE assumed that all drifts in the repository are intersected by the dike(s), magma fills all drifts, and all waste packages in the repository are damaged but remain in the drifts. No waste is released directly into the accessible environment in an intrusive igneous event, but radionuclides are released to the accessible environment through subsequent groundwater transport. DOE models this transport to occur through the same pathways represented in the nominal, seismic and early failure scenario classes, which are evaluated in TER chapters on unsaturated zone flow and transport (Sections 2.2.1.3.6, 2.2.1.3.7, 2.2.1.3.8, and 2.2.1.3.9). During an extrusive, or volcanic, igneous event, DOE considered that magma continues to rise to the surface as a dike after possibly intersecting repository drifts and, on the basis of the behavior of basaltic eruptions in general, that surface activity along the resulting initial fissure would rapidly localize, or focus, to a single, or few, points of effusion (SNL, 2007ae; SAR Section 2.3.11.4.1). A wider volcanic conduit would be expected to develop at that focus somewhere along the dike by excavation from the surface vent downwards. This conduit can potentially intersect a drift(s) or develop in the area between the drifts. Magma flow up a drift-intersecting conduit entrains waste from disrupted packages, thereby providing a direct pathway for waste material to be released to the accessible surface environment during a volcanic eruption.

DOE explained that the volcanic (extrusive) part of the igneous scenario is an extension of the intrusive part (SAR Section 2.3.11.1) and concluded that every intrusive event that might intersect the repository is likely to have a conduit develop somewhere along one of the dikes, as described in SAR Section 2.3.11.2.1.2 and SNL Table 7-1 (2007ae). The conduit (or conduits) may, however, form outside the repository footprint or may not intersect a drift, and in that case, no waste material would be entrained into the magma that rises to feed the eruption. In effect, this would be equivalent to the intrusive-only case. In addition, DOE determined that conduits that might feed surface volcanoes may only develop along specific parts of dikes (SAR Section 2.3.11.4.2.1) and thus concluded that the probability of a volcanic event occurring at the repository is expected to be lower than the probability of an intrusive event. DOE also concluded that if an eruption that entrained waste material and transported it into the surface environment did occur at the repository, the potential doses to the RMEI location from radionuclides released through the intrusive and extrusive pathways would be additive. Further details of conduit development are evaluated in the NRC staff's review of the volcanic eruption modeling case (TER Section 2.2.1.3.10.3.3).

2.2.1.3.10.3.2 The NRC Staff's Review of DOE Igneous Intrusion Modeling Case

The DOE model for igneous intrusion and its effect on repository performance rely on four key conclusions:

- Magma rising as a dike beneath repository drifts will intersect and flow into the drifts.
- Any dike intersection into the repository footprint floods all drifts with magma, causing engineered barrier system components, including all waste packages and drip shields, to fail while magma and waste remain in the drift.
- Igneous intrusion does not alter the ambient hydrologic flow and transport regime significantly (i.e., the natural barriers above and below the drifts are not affected).
- Subsurface conduits that develop beneath volcanoes can be represented by cylinders and only entrain waste within the part of the cylinder that intersects the drift.

Conclusions 1–3 solely concern the intrusive model case, while Conclusion 4 is also applicable to the volcanic model case. NRC staff's review focuses on the risk-significant aspects of these conclusions. NRC staff's overall risk perspective for these abstractions is given in the next subsection, and the specific technical aspects are evaluated under the subsequent subsections.

NRC Staff Perspective on Risk

NRC staff has assessed the risks caused by an igneous event at the proposed repository on the basis of DOE's information. As stated in the Introduction section of this chapter, while the probability of an igneous event is low, the consequences could be potentially high. The igneous intrusion modeling case would constitute most of the calculated dose for the first 1,000 years following permanent closure of the repository, as shown in SAR Figure 2.4-18(a), and is approximately half the calculated dose for the seismic ground motion modeling case in the ensuing 9,000 years. However, for the first 10,000 years, SAR Figure 2.4-18(a) indicates that the mean annual dose from igneous intrusion is on the order of 0.001 mSv [0.1 mrem]. Moreover, in SAR Section 2.1, DOE indicated that for the post-10,000-year period, the igneous intrusion modeling case and seismic ground motion modeling case provide approximately equal contributions to the total mean annual dose to the reasonably maximally exposed individual (RMEI) for the last 300,000 years of the time period. SAR Figure 2.4-18(b) shows that the mean annual dose for each modeling case is on the order of 0.01 mSv [1 mrem] in the post-10,000-year period.

In SAR Section 2.4.2.2.1.2.3, DOE provided the probability-weighted consequences of igneous activity (intrusive and extrusive) using the probability distribution from its expert elicitation for a Probabilistic Volcanic Hazard Assessment (PVHA). DOE identified that the probability-weighted igneous mean intrusive dose is estimated to be less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period and the median dose is less than 0.005 mSv/yr [0.5 mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.1). DOE estimates for the probability-weighted igneous extrusive (volcanic eruptive) mean dose alone are on the order of 10^{-6} mSv/yr [0.0001 mrem/yr] for the 10,000-year period, and the median dose is less than 6×10^{-7} mSv/yr [6×10^{-5} mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.2). The NRC staff recognizes that the difference in magnitude for the dose consequences between the two igneous scenarios (intrusive and extrusive) predominantly results from the different number of waste package failures estimated to occur for each scenario, which, from a risk perspective, causes the dose from the extrusive case to be relatively low (SAR Section 2.2.1.4.1).

Effects of Igneous Intrusion on Performance of Natural Barriers

Because DOE did not rely on an evaluation of igneous events in its demonstration of multiple barriers, the NRC staff does not include a discussion of igneous events in TER Section 2.2.1.1. DOE screened out of its performance assessment the effect of igneous dikes and sills on groundwater flow and transport pathways surrounding drifts in the upper and lower natural barriers, as described in SNL (2008ac) (FEP 1.2.04.02.0A). At the drift wall, however, DOE included the effect of igneous intrusions, by assuming the drifts become degraded and the seepage barrier is eliminated. For this case in the performance assessment, seepage is set equal to percolation. Igneous activity near repository drifts may alter hydrologic properties of the host rock or cause perching of water in the unsaturated zone. DOE's sensitivity analyses indicate that these effects on unsaturated zone flow in repository performance are small (FEP 1.2.04.02.0A; SNL, 2008ac). In particular, the potential effect of increased fracturing in and around a dike providing preferred water pathways has relatively little impact, given the predominance of fracture flow in the existing, undisturbed unsaturated zone beneath much of the repository footprint (see the NRC staff review of unsaturated zone flow in TER Section 2.2.1.3.6). Farther into the far field, igneous dikes and sills may modify saturated zone flow and plume pathways, but again, DOE suggested these effects to be minor for performance (FEP 1.2.04.02.0A; SNL, 2008ac).

The NRC staff reviewed relevant information in the SAR (Section 2.3.11.2.1.1), in SNL (2008ac), and in SNL (2007ag) and notes that the DOE treatment of igneous activity in the form of dikes and sills is reasonable with respect to groundwater flow in the natural barriers on the basis of the following rationale. For dikes and sills that

- Intersect drifts, the seepage barrier is eliminated from DOE's performance assessment model

- Occur above the repository, any potential increases in focusing of groundwater flow that might be caused by the presence of intrusions (dikes and sills) are not important because, as noted in TER Section 2.2.1.3.6.3.2, that uncertainty in spatial variation in groundwater percolation (flow) is unimportant to performance

- Occur below the repository in the unsaturated zone, sensitivity analyses for groundwater transport that included potential changes to hydrologic properties of fractures and matrix lead to a smaller effect than that already considered for the uncertainty in net infiltration (FEP 1.2.04.02.0A; SNL, 2008ac)

- Occur in the saturated zone, the area the igneous activity and resulting rock bodies can potentially modify is small compared with the size of the saturated flow zone, with typically only a few decimeters [4–12 in] of disrupted rock around the ~1-m [3 ft]-wide dikes (SAR Section 2.3.11.3.2; SNL, 2007ag; Detournay, et al., 2003aa; Keating, et al., 2008aa); moreover, the probability of an intrusion occurring directly within the groundwater plume is also small.

In the igneous intrusion abstraction, DOE ignored sills for the reasons given in TER Section 2.2.1.3.10.3.1 and because DOE's performance assessment assumes that a dike intruding the repository would result in failure of all waste packages and drip shields. The NRC staff notes that a potential sill would intersect fewer drifts than a dike swarm, and, therefore, DOE's dike model encompasses the potential consequences of sills. The NRC staff notes that DOE's approach to sills in its igneous intrusive scenario captures the potential impacts on repository performance.

Behavior of Intruding Magma in Drifts and Effects on the Engineered Barrier System

In developing the model for subsurface igneous processes, DOE concluded that basaltic magmas in the Yucca Mountain region would contain appreciable amounts of dissolved volatiles, primarily water. This dissolved water would form a gas phase as pressure on the magma becomes lower due either to normal ascent toward the surface or by intersection with a repository drift (SAR Section 2.3.11.2.1.2). Significant amounts of gas expansion in the upper 300 m of rise [above ~1,000-ft depth] would cause magma in potential igneous events to flow more rapidly, and perhaps more extensively, than would be expected for magmas with little gas-driven expansion. In part because of the relatively high dissolved water contents expected for Yucca Mountain basaltic magmas, DOE concluded that all repository drifts would be rapidly filled by magma flow if a future intrusive igneous event occurred within the repository footprint (SAR Section 2.3.11.2.1.2).

The NRC staff reviewed the information DOE presented to support the conclusion that basaltic magmas are expected to have relatively high dissolved water contents. The NRC staff notes that information DOE cited supports this conclusion (SAR Section 2.3.11.3.2.3), and, in addition, the presence of hydrous minerals in some Yucca Mountain region basaltic rocks supports the DOE conclusion that dissolved magmatic water contents in these magmas are relatively high (Nicholis and Rutherford, 2004aa; SNL, 2007ae). The NRC staff notes that uncertainties in estimates of the dissolved volatile content of these basalts do not affect performance significantly, given that DOE assumes that magma behavior at repository depths is driven by an exsolved gas phase.

In the DOE intrusive igneous case, DOE assumes that if a single rising dike intersects any part of the repository footprint where drifts containing waste packages are located, then all drifts in the repository are rapidly filled with magma. DOE developed this approach to account for the uncertainties in determining the physical characteristics of dikes at repository depths and for uncertainties in magma flow processes in drifts intersected by dikes (SAR Section 2.3.11.3.1). For the ascending magma entering the drifts, DOE recognized that there are two end member possibilities for flow behavior, considering how rapidly and violently magma could enter a drift. The less rapid end-member is termed effusive, as in a lava-like flow, while the other is more explosive, resulting in a fragmental, or pyroclastic, flow (SAR Section 2.3.11.2.1.2; SNL, 2007ag; Woods, et al., 2002aa; Dartevelle and Valentine, 2005aa, 2009aa). The NRC staff also conducted independent confirmatory investigations (e.g., Woods, et al., 2002aa; Lejeune, et al., 2009aa) verifying that potential magma flow into drifts could occur quickly enough so that only minor cooling of the magma would occur. On the basis of the results of these independent studies and its own evaluation, NRC staff notes that DOE has developed a reasonable technical basis to propose that all drifts will be filled with basaltic magma if an intrusive igneous event occurs at the repository site. Because this approach involves the disruption of all waste packages stored in the proposed repository (SAR Section 2.3.11.3), as explained next, the NRC staff further notes that this does not underestimate risk and that there are no technical uncertainties in this aspect of DOE's approach that could reasonably increase the DOE risk estimates.

According to DOE's calculations, after intersection and intrusion by magma drift temperatures are modeled at or near magmatic temperatures of 1,046–1,169 °C [1,915–2,136 °F], at which point plastic deformation of the waste packages begins. Additional DOE analysis showed waste packages could also be damaged by magmatic pressures as low as 4 MPa [580 lb in^{-2}]. DOE concluded waste package failure could result in waste forms that are exposed to high temperatures and that undergo chemical reactions with magma and its constituents. DOE assumed that the packages would encounter additional mechanical loads from the cooling and solidification of enveloping magma. Already weakened by the thermal effects of the magma, the mechanical loads associated with the magma would result in deformation of the waste package. DOE proposed that similar effects would occur for drip shields exposed to magmatic conditions. Thus DOE concluded that uncertainties associated with the potential effects of magma on waste package and drip shield performance are sufficient to warrant the assumption that all waste package and drip shield barrier capabilities are removed in models for igneous intrusive events (SAR Section 2.3.11.3.2.4). DOE also concluded that exposure to magmatic conditions will result in unprotected waste forms that are, effectively, instantaneously degraded, such that radionuclides are assumed to be immediately available for hydrologic transport, as soon as the intrusion is cool enough to allow water to contact waste (SAR Section 2.3.11.3.2.4). As discussed in the next subsection, the cooling time of the intrusion is short relative to the time scale of groundwater percolation and flow, and relative to the period for postclosure repository performance. DOE further concluded that although the waste package no longer serves as a barrier to water flow after an igneous event, corrosion products from degradation of waste package materials will be present and will strongly retard release of certain radionuclides into the unsaturated zone, in the same manner as in the nominal scenario. The NRC staff evaluates the role of corrosion products in radionuclide release in TER Section 2.2.1.3.4.

The NRC staff reviewed the information DOE provided in SAR Section 2.3.11.3.2 to support the representation of engineered barrier and waste form responses to potential igneous intrusive events. The NRC staff notes that the DOE approach for modeling waste package and drip shield response to potential intrusive events is reasonable, because this approach is consistent with available information on the heat of the magmatic intrusion and possible behavior of the proposed type of waste packages, drip shields, and waste form, as discussed in the previous paragraphs. Further, given that DOE's analysis fails the engineered barriers for all waste packages, staff has identified no reasonable alternatives to this approach that would result in higher calculated doses. The NRC staff notes that the DOE representation of waste form response as instantaneously degraded with all radionuclides available for subsequent hydrologic transport is reasonable, as this approach is consistent with available information listed previously, and there are no reasonable alternatives that would result in higher calculated doses.

Because DOE assumes that all waste package and drip shield capabilities cease during an igneous intrusive event, there are few uncertainties that are significant to the evaluation of potential intrusive igneous events. Those that DOE proposed to have potential significance to dose are discussed in the next two sections. The NRC staff has determined, by reviewing DOE-provided information, that the physical conditions following magma intrusion into a drift could affect subsequent hydrologic (groundwater) flow and transport processes. Thus, the remainder of this section evaluates the DOE basis for calculating the effects of magma cooling on the drift environment and subsequent hydrologic flow and transport.

Magma Cooling and Heat Flow to Host Rock

The temperature in the drifts after magma intrusion is an output parameter to TSPA (SAR Section 2.3.11.6.7). This subsection evaluates the DOE estimates of centerline and wall temperature in the invaded drifts in the period after intrusion and the timing of the intrusive event with respect to the repository life cycle, reflecting the temperature of the host rocks during the period of heating by radionuclide decay. DOE included these for consideration of the post-intrusion environment in the damaged drifts and the period of time after which groundwater seepage through the drifts could return.

The temperature in the drifts after magma intrusion provides an estimate of the cooling time of the basalt inside the drift. The cooling of the basalt inside the drift and the drift centerline temperature, as well as drift-wall temperature, also influence the spent fuel dissolution model and the calculation of diffusion coefficients. Diffusion coefficients are used to calculate near-field contaminant transport in the unsaturated zone rock.

DOE concluded that a short-lived (hours to days) intrusive event, as in SNL Figure F–1 (2007ab), would fill every drift in the proposed repository with basaltic magma at a temperature of approximately 1,100 °C [2,012 °F]. Following the intrusive event, the magma in the drifts begins to cool. DOE performed numerical simulations to model the magma cooling and heat flow in the rock between drifts, via non-steady-state heat conduction, with radial flow of heat from the magma-filled drifts into the host rock. DOE's model considers a single basalt-filled drift, recognizing that the heat from one 5-m [16-ft]-diameter magma-filled drift will not influence the next drift approximately 80 m [262 ft] away (SNL, 2007ag). The calculated temperature decreases with time and distance from the centerline of the drift. The thermal diffusivity of the rock is calculated using the rock volumetric heat capacity and the thermal conductivity of the welded tuff at the repository horizon. The host rock in the heat-flow calculation is assumed to be either completely dry or completely wet. Thermal diffusivity of the welded tuff and the basaltic magma are assumed to be the same. The drift wall temperature prior to magma intrusion in the DOE model runs was between 25 and 200 °C [77 and 392 °F]. DOE concluded that this range suitably represents temperatures at different times for the intrusive event (reflecting elevated repository temperatures for several thousand years after closure (e.g., see SAR Figure 2.3.5-33 for calculated repository drift-temperature decay curves). These temperature distributions provided the DOE estimate of the cooling rate and thermal history of the repository and the drifts following an intrusive event.

DOE identified that the model does not include the effects of latent heat of magma crystallization or the property contrasts between the magma and the tuff. Without latent heat effects, the one-dimensional model results underestimated peak temperatures and time needed for cooling. Therefore, DOE considered alternative models, including an analytical solution that approximated the effects of latent heat and numerical solutions in two dimensions that included both latent and radioactive heat. Noting that latent heat would be liberated during magma crystallization and that its effects would be most pronounced at very early times while the magma is still partially liquid, DOE accounted for the effect of latent heat by increasing the initial temperature of the magma.

DOE considered the main uncertainty when modeling magma cooling and solidification to be the initial magma temperature. For dry magma, 1,150 °C [2,102 °F] was used, but the NRC staff notes that magma with a high water content could have a temperature as low as 1,046 °C [1,915 °F] (Nicholis and Rutherford, 2004aa). Although the difference is small, the NRC staff determined that DOE stated that this effect could slightly reduce the time required for the magma to cool to a solidified rock. However, as noted previously, DOE assumed a higher initial magma temperature instead of explicitly including latent heat of crystallization. Other uncertainties DOE considered included thermal conductivity, grain density, specific heat capacity, matrix porosity, saturation, and the lithophysal porosity of the host tuff. Heat loss was modeled as purely conductive, as DOE did not expect convection to occur in stagnant magma within drifts. For an igneous intrusion event occurring after about 1,000 years into the preclosure period, DOE concluded that the repository drift walls would attain a temperature of 100 °C [212 °F] about 100 years after the intrusive event occurs, as in SNL Figure 2.3.5-33 (2008ag).

DOE showed that drift temperatures in the 100-year post-intrusion period abstracted to the Total System Performance Assessment (TSPA) have little influence on dose estimated from the intrusive scenario. The NRC staff notes this is reasonable, because DOE's adopted scenario involves the disruption of all waste in the repository and because potential changes to dose from temperature increases due to magmatic heat are short lived (less than or equal to 100 years) compared with the time scale of groundwater percolation and flow, and relative to the period of postclosure repository performance. Also, as explained in the next subsection, some radionuclides that are important to dose have an inverse solubility relationship with temperature. NRC staff also notes that DOE's conclusion that cooling basaltic magma would not significantly affect drift characteristics that are relevant to drift degradation is reasonable, again because DOE's adopted scenario involves the disruption of all waste in the repository. DOE used the final temperature of the drift and the cooled basalt temperature as an input to calculate the spent fuel dissolution model and diffusion coefficients. While the latent heat of crystallization would result in a slightly longer cooling time for the basaltic magma while it was still partially liquid, NRC staff notes that this would be offset by the DOE assumption of a slightly high initial magma temperature. However, DOE concluded that an extended magma cooling time would have little influence on the dose estimated from the intrusive scenario, and on the hydrologic flow and transport after an igneous event, which the NRC staff also notes is reasonable because of the short-lived cooling time for an intrusion compared with the much greater timescale for groundwater flow.

Percolation Flux Through Cooled Basalt

Chemical changes, expressed as the pH and ionic strength, to groundwater that may react with the new basalt rock filling repository drifts after a future intrusive magmatic event is an output parameter to TSPA (SAR Section 2.3.11.6.7; SNL, 2005ae). This subsection evaluates DOE estimates of possible chemical changes that might occur to groundwater as it begins to seep through and possibly react with cooling, and cooled, basaltic material filling the drifts.

In considering percolation of groundwater through the drift after an igneous intrusion into the repository, DOE assumed that solidified basalt rock in the drift has the same fracture, porosity, and permeability characteristics as the surrounding tuff. DOE also concluded that the newly introduced basalt rock could affect the chemistry of water that seeps into the drift; in particular, pH and ionic strength. To examine possible changes in these two chemical parameters of the seepage water, DOE selected for numerical analysis three groundwater samples from large, fractured basalt-hosted reservoirs and conducted an extensive literature review of the chemistry of basalt-hosted waters to provide a range of pH and ionic strength values, as described in SAR Sections 2.3.7.5.3.1 and 2.3.11.3.2 and SNL Section 4.1.2 (2007ae). Temperature can affect the pH of incoming fluids, so to avoid underestimating radionuclide solubilities, DOE calculated the parameter values at 25 °C [77 °F], rather than at higher temperatures that would have resulted in lower solubility limits (radionuclides of concern show retrograde solubility in this pH range) and therefore smaller mass releases.

As discussed in the previous subsection, for an igneous intrusion event occurring approximately 1,000 years into the postclosure period, water seepage and flow through the host rock mass is estimated to resume about 100 years after an intrusion occurs. This is equivalent to the time when the basalt in the drifts would reach ~100 °C [212 °F] along the drift centerline (SAR Section 2.3.11.3.3.8; SNL, 2008ag). This time also corresponds to when the repository drifts walls are assumed to cool below the local boiling temperature, as shown in SNL Figure 2.3.5-33 (2008ag). DOE modeled reestablishment of groundwater percolation through the invaded repository drifts and failed engineered barriers. DOE did not model release of radionuclides in gaseous form from the waste packages, because DOE's analyses indicate that this does not influence the final dose at the receptor (SNL, 2008ag). Hence, groundwater percolation was the only pathway DOE considered for release of radionuclides. The NRC staff notes that the gaseous releases from a potential intrusive event would have very limited impact on overall calculated dose and that the groundwater pathway would be dominant.

DOE's review and analysis of relevant information on basalt-hosted groundwater, as a proxy for water entering a cooled, intruded drift, showed that pH and ionic strength of water prior to entering basalt reservoirs, as well as variations within the actual composition of the basalt, are likely to have little effect on the pH and ionic strength of the water exiting the invaded drifts. DOE's sensitivity analyses using waters from basaltic aquifers also showed that the liquid influx composition has an insignificant effect on the in-package chemistry model estimates. DOE analyses using waters equilibrated with the ambient-temperature Columbia River Plateau and Iceland basalts also found that the pH and ionic strength of the incoming water would have little influence on the resulting pH and the ionic strength after water has passed through the basalt-filled repository.

The NRC staff notes that the Columbia River Plateau and Icelandic basalt compositions encompass a wide range of basalt types that include the characteristics of basalts at Yucca Mountain and are thus a satisfactory proxy for expected groundwater compositions in the case of outgoing (effluent) groundwater flow after passing through basalt-filled drifts from an intrusive event at the proposed repository. This is because the basalt rock types in those provinces encompass the same compositional ranges as expected for a future basaltic igneous event at Yucca Mountain (SNL, 2007ae). Thus, DOE's analyses of uncertainties associated with the expected composition of future repository-filling basalt suggest that the uncertainties would not significantly affect the chemical composition of the effluent groundwater (SAR Section 2.3.11.3.3.9). In this case, the modeled changes in groundwater chemistry after contact with a basalt-filled drift would be negligible. This is consistent with the relatively small volume of intruded basalt in comparison with the volume of host rock, and that, even following an intrusive event, the effluent groundwater composition would be dominantly controlled by reaction with the contents of the failed waste package (i.e., spent fuel or high-level waste glass, and corrosion products from internal components) rather than the intruded basalt, as described in SNL Section 6.8.10 (2007ae). The NRC staff review of release and transport of radionuclides following a possible igneous intrusion is further detailed in TER Section 2.2.1.3.4 (specifically, Sections 2.2.1.3.4.3.2 on waste form degradation and 2.2.1.3.4.3.4 on colloid formation and stability).

Summary of NRC Staff's Review of the Igneous Intrusion Modeling Case

The NRC staff notes that DOE has provided information to support the modeling approach used to represent intrusive igneous events in the performance assessment of the proposed repository. The NRC staff notes, on the basis of its evaluation of DOE-provided information, that DOE has considered how cooling basaltic magma would affect the characteristics of the invaded and disrupted repository drifts that are relevant to hydrologic flow and transport and to the calculated dose. The DOE modeling approach relies on the assumption that intersection of any igneous intrusive feature into the repository footprint fills all of the repository drift with basaltic magma and that the magma removes all barrier capabilities from all waste packages and drip shields in all drifts. This assumption is reasonable, because it would not underestimate the risk from a potential intrusive igneous event. Moreover, the NRC staff notes that DOE has evaluated the uncertainties associated with this assumption that could increase the DOE dose estimate. DOE has represented the potentially significant effects of igneous intrusive events in the performance assessment. The igneous intrusion case provides insight into repository performance in the case of failure of engineered barrier components (drip shields and waste packages). It shows that the technical basis for the capability of those barrier components is based on and consistent with the technical basis for the performance assessments (reviewed in TER Section 2.2.1.4.1).

2.2.1.3.10.3.3 The NRC Staff's Review of DOE Volcanic Eruption Modeling Scenario

DOE concluded that in all potential igneous intrusive events that intersect the repository footprint, a rising dike would reach the surface and develop a conduit at some location along the intrusion and magma would be extruded. If a conduit is located wholly or partially in a repository drift, waste from disrupted waste packages could be entrained by magma flow up the conduit and erupted from a volcano at the surface. Compared with the intrusion scenario, in which the contents of all waste packages in the repository are made available for hydrologic transport, DOE concluded that, for the volcanic scenario, only a limited amount of high-level waste could be entrained directly into a conduit or conduits (SAR Section 2.3.11.4).

In the type of basaltic volcanic activity considered by DOE for a future eruption through the repository, a dike reaches the surface and activity begins along a fissure (an elongated system of vents, which is the surface expression of the dike; see SAR Sections 2.3.11.2.1 and 2.3.11.4.1.1 and SAR Figure 2.3.11.5). In DOE's model, magma flow to the surface in the dike usually localizes to a single, or a few, points over a period of hours to a few days, as observed at past basaltic eruptions and previously discussed in TER Section 2.2.1.3.10.3.2. Such behavior was seen in analog historic events [e.g., the 1943-1952 eruption of Paricutín, Mexico; the 1973 Heimaey eruption in Iceland; and the 1975 Tolbachik eruption in Kamchatka (Pioli, et al., 2008aa; Thorarinsson, et al., 1973aa; Doubik and Hill, 1999aa)]. DOE studies of igneous products exposed in the rock record also inferred a similar style for some prehistoric basaltic eruptions (e.g., SAR Section 2.3.11.4; SNL 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa). At this point in the modeled eruption, a conduit is considered to develop below the point of localization, with the main vent at the surface. This conduit feeds an explosive and lava-flow-forming Strombolian-style eruption. DOE adopted a violent Strombolian style for the entire model eruption considered, on the basis of the characteristics of the young Lathrop Wells scoria cone near Yucca Mountain (see TER Section 2.2.1.2.2). DOE recognized that conduits grow (widen) downwards from the surface in the plane of the dike, as detailed in SAR Section 2.3.11.4.2.1.2 and SNL p. 6-46 (2007ae), and thus, in DOE's repository-disruption scenario, intersect a drift through the roof.

DOE characterized subsurface volcanic conduits as flaring inward down from the top of the surface vent, such that conduit diameters at repository depths will be smaller than those observed near the surface. DOE characterized the size and shape of conduits using studies at exposed analog volcanoes (e.g., SAR Section 2.3.11.4 and Figure 2.3.11-6; SNL, 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa) and theoretical considerations and model studies (e.g., Wilson and Head, 1981aa; Valentine, et al., 2007aa). In the performance assessment, DOE represents subvolcanic conduits as simple cylinders (SAR Section 2.3.11.4.1). DOE used the area of the conduit that intersects a drift to calculate the mass of waste the conduit entrains. DOE concluded that entrained waste is mixed uniformly in the volume of magma that is subsequently erupted at the surface.

From a risk-perspective, the DOE performance assessment calculates that the expected annual dose from the igneous volcanic modeling case alone is approximately 0.1 percent of the dose calculated for the intrusive scenario (SNL, 2007ag). This difference between the volcanic and intrusive scenarios arises, in part, because DOE concluded that the volcanic scenario entrains and erupts approximately 0.1 percent of the amount of high-level waste that is disrupted during the intrusive case. Thus, the NRC staff's review of the subsurface processes associated with the volcanic case focuses on the DOE basis for concluding that a volcanic conduit, or conduits, would entrain a limited amount of waste.

Development of Conduits and Likelihood of Ejecting Waste in a Volcanic Eruption

In the DOE-developed model, one to three eruptive conduits may occur along the thickest dike. DOE treats the predicted location of a single conduit along a dike, the most likely occurrence, as random (SAR Section 2.3.11.4.2.1). In SAR Section 2.3.11.2.2, DOE developed a basis to determine the likelihood that at least one conduit will form through the repository footprint and, more specifically for risk significance, through an emplacement drift containing waste packages if a dike intersected the repository. The DOE model for conduit formation is based on observations at basaltic volcanoes and supported by calculations constrained by information obtained from studies of analog eroded volcanoes (SNL, 2007ae).

On the basis of observations of Quaternary volcanoes in the Yucca Mountain region, where mostly only one volcano develops along a dike (Keating, et al., 2008aa), DOE heavily weighted the distribution of the likely number of conduits that might develop along a dike toward one conduit per eruption (SAR Section 2.3.11.2.1.2; SNL, 2007ae), and in this way treated uncertainty. DOE determined that the presence of repository drifts would not affect the rise of a dike, nor subsequent eruptive processes, because the drifts would be negligible in volume compared to the volume of rock the dike transects. The NRC staff notes that this assumption is reasonable, given the expected small size, on the order of a 1 to 2 m [3 to 6 ft] width below repository depths (Keating, et al., 2008aa), and energy of a propagating dike. DOE determined that 85 percent of past eruptive events have formed a single conduit, 10 percent formed 2 conduits, and 5 percent formed 3 conduits, and this same information also suggests that multiple conduits should be spaced between 0.4 and 2 km [0.25 and 1.2 mi] apart. DOE also considered five alternative conceptual models to represent the location of a conduit along a dike. On the basis of field analogs, models, and studies presented in SNL (2007ae, 2007ag), DOE concluded that a model for random location of a conduit along an existing dike is the only supportable approach, because conduits do not have any predictable location along surface expressions of dikes in analog examples (Doubik and Hill, 1999aa; Hill and Connor, 2000aa; Valentine, et al., 2006aa; Valentine and Krogh, 2006aa; Keating, et al., 2008aa).

To calculate the likelihood that at least one volcanic conduit will form through an emplacement drift and entrain waste, DOE used numerical models to simulate the number of dikes that could penetrate the repository footprint, using dike characteristics from CRWMS M&O (1996aa). For each simulation, DOE calculated the length of the dike, or dikes, located inside and outside the repository footprint and found there was a 60 percent chance that more than one dike would form in an event. For the widest dike in each simulation, DOE constrained its model to form one to three conduits at random locations along that dike and determined whether this location coincided with the repository footprint (SAR Section 2.3.11.4.2.1.3; SNL, 2007ar). Using this approach, DOE estimated that there was a 20 to 35 percent chance, with a mean of 28 percent, that at least one conduit would form within the repository footprint. This value reflects the relatively small size of the repository footprint in comparison with the total area that dikes could impact (SAR Section 2.3.11.2.2). On the basis of alternative volcanic event characteristics and behavior, DOE acknowledged that the conditional likelihood of at least one eruptive center (conduit) within the repository footprint might range from 43 to 78 percent (SAR Section 2.3.11.2.2.6). However, DOE concluded that, on the basis of features of Yucca Mountain repository volcanoes, a mean conditional eruption probability of 0.28 (28 percent) times the probability of dike intersection with the repository footprint was most consistent with basaltic volcanic events that are expected to include multiple dikes and in which conduit(s) form on the widest dike. On this basis, the mean conditional probability of a conduit forming within the repository, using the mean intrusive probability from the PVHA expert elicitation of 1.7 × 10^{-8} per year (SAR Section 2.3.11.4.2.1; TER Section 2.2.1.1.3), is 4.8 × 10^{-9} per year.

The 28 percent conditional factor DOE provided is for a conduit that develops within the repository footprint, but which may not necessarily eject waste. DOE then developed a second conditional probability, given as 0.296 (NRC staff rounded this to 0.3, or 30 percent), to represent the fraction of conduits within the repository footprint that may actually intersect a drift containing waste packages and eject the waste contents through a volcanic vent (SAR Section 2.3.11.4.2.1). This factor accounts for the spatial distribution of waste emplacement drifts within the repository footprint area and the likely orientation of dikes.

The staff reviewed DOE information regarding the likelihood for conduit development at repository drifts. Using its knowledge of the characteristics of basaltic volcanism at the Yucca Mountain region and DOE and independent confirmatory studies of conduit development in basaltic volcanism (BSC, 2003ab; Detournay, et al., 2003aa; Hill and Conner, 2000aa; Pioli, et al., 2008aa), the NRC staff determines that DOE has reasonably characterized the number and spacing of volcanic conduits. The DOE conclusion that the processes leading to conduit development along a dike are reasonably represented as randomized along the widest dike segment is reasonable, because available information shows that there is no predictable pattern controlling conduit formation at other analogous basaltic volcanoes. The NRC staff reviewed the DOE methodology that developed the 28 percent factor for conduit development in the repository and the 30 percent factor for conduit intersection with a drift. DOE implemented randomized conduit development in developing these factors. NRC staff notes that even if the conduit development factor was significantly higher, the implied risk would change by only a small amount (e.g., using a factor of 100 percent would increase the amount of waste disrupted and ejected to ~0.3 percent of that disrupted in the intrusive case). Given the relatively small volume and rapid infilling time of the intersected drifts, NRC staff notes that the presence of repository drifts will not significantly affect the localization process for conduit development. Thus, NRC staff notes that DOE has evaluated the likelihood of conduit development at intersected drifts.

Eruptive Conduit Growth and Size, and Impact on Waste Packages and Waste

According to DOE's scenario presented in SAR Section 2.3.11.4.2.1, one or more conduits may intersect repository drifts, all the waste packages within the area of the conduits are assumed to be destroyed, and all the waste is assumed to be incorporated into the erupting magma (SNL, 2007ag). The waste is assumed to mix with magma and be carried up the conduit toward the surface, where the magma–waste mixture would be explosively ejected into the atmosphere or flow as lava along the ground.

DOE considered the failed waste packages directly intersected by a conduit to provide no protection against waste release, so in the DOE model, the conduit size at repository depth directly determines the number of waste packages disrupted. More specifically, DOE calculated the number of waste packages intersected by conduits as a cumulative distribution function that is based on a distribution for the number of conduits, a distribution for conduit diameters, and the likelihood factors for location of the conduits on the dikes, which includes the design configuration of the subsurface repository. Accordingly, DOE considered additional parameters including waste package size and spacing, drift location and dimensions, and distributions for dike length, orientation, thickness, and number of dikes in an intrusive event. DOE concluded that rising magma in a dike that enters a drift will slow relative to that in solid rock pillars between drifts; thus the dike segment above drifts will lag slightly in breaching the surface.

From that conclusion, DOE proposed that vents and conduits are more likely to form between drifts than above them. In most realizations tested by DOE, this led to a condition where the volcanic conduit forms along the dike in the rock pillars between drifts and not the drift itself; thus the most likely value for the number of disrupted waste packages in the model is zero. From zero to seven waste packages were modeled in the Total System Performance Assessment (TSPA) as intersected by a conduit during an eruption.

In DOE's model, uncertainty in conduit size is bounded by a size distribution based on observed host-rock fragments in violent-Strombolian deposits at Lathrop Wells volcano (Doubik and Hill, 1999aa) and in SNL Section 6.4 and Appendix C (2007ae) and on field studies at analog sites, which DOE interpreted as suggesting that the diameter is largest at the surface and decreases with depth. DOE gave a distribution for conduit diameters from approximately 4 m [13 ft] (bounded by dike width) to a mean value of 15 m [50 ft] and a 95th percentile value of 21 m [69 ft] for an expected conduit diameter at repository depth (SAR Section 2.3.11.4.2.1.2), with DOE's volcanic scenario analysis conduits developed only where the trend of a dike intersected a drift (SAR Section 2.3.11.4.1.1.1). DOE concluded that it is highly unlikely that a secondary conduit will form at some point along the drift away from the dike intersection. This conclusion was based on DOE's view that magma will solidify quickly and pressures will be insufficient to allow the formation (or maintain the opening) of a secondary dike, fed from the magma in the drift. In the analysis involving pyroclastic flow of magma inside a drift (an alternative conceptual model mentioned previously with respect to the intrusive case), DOE assessed one situation where it assumed that a secondary fracture had already formed and a secondary opening was created on the drift-top wall (BSC, 2005af). DOE applied a multiphase fluid dynamics analysis to this scenario. Simulated results exhibited intermediate behavior with a down-drift multiphase flow on the roof and a return flow on the floor. The whole system with these two openings formed a clearly defined recirculation pattern in the drift with some materials leaving the system and some materials recycling back into the drift along the roof. Simulations also showed that this scenario leads to relatively high dynamic pressures compared with a single-conduit situation. Other simulations indicated that blockage of the volcanic conduit might also create secondary breakouts at a point away from the location of initial dike intersection (SAR Section 2.3.11.3.2.2). Although DOE acknowledged that the chance of these scenarios occurring was unlikely, it concluded that such scenarios could lead to a one to two order-of-magnitude increase in the amount of waste released during a volcanic igneous event (essentially equivalent to the waste content of a single drift, ~70–100 waste packages), which would cause no more than a one to two order-of-magnitude increase in expected annual dose (SAR Section 2.3.11.3.2.2).

The NRC staff's review of the DOE approach to modeling the development of subvolcanic conduits is guided by the information in SAR Section 2.3.11.3.2.2 regarding dose sensitivity to the waste source term and analyses in SAR Section 2.4.2 showing the volcanic case contributes approximately 0.1 percent of the total dose for the igneous scenario. Staff determined that the DOE mean conduit diameter of 15 m [49 ft] (SAR Section 2.3.11.4.2.1.2) appears to be influenced by estimates of the conduit diameter for Lathrop Wells volcano; the calculation for the conduit diameter given in SNL Appendix F (2007ae) and adopted in the SAR appears to be in error. This diameter was recalculated using the same information by Valentine, et al. (2007aa) at ~8 to 9 m [~26 to 30 ft], which NRC staff notes is the correct value. In contrast, many of the smaller conduit diameters DOE used in supporting this parameter value are from eroded volcanoes in the Yucca Mountain region that DOE concluded are not representative of expected basaltic igneous processes (e.g., CRWMS M&O, 1996aa). On the basis of this information and other published information about basaltic volcanic conduits located several hundred meters [~1,000 ft] below surface (e.g., Doubik and Hill, 1999aa;

Valentine and Groves, 1996aa; Valentine and Krogh, 2006aa; Delaney and Gartner, 1997aa), the NRC staff notes that uncertainty in the average and maximum conduit diameter may be up to a factor of five greater than DOE considered. The magnitude of this uncertainty, however, would increase the expected annual dose for volcanic igneous events by less than an order of magnitude. Because DOE calculates that the volcanic case contributes 0.1 percent of the total dose to the igneous scenario, NRC staff notes that this increase in uncertainty in conduit diameter, and thus dose, would not be significant. Thus, the DOE approach for representing subvolcanic conduits is reasonable.

Evaluation of Magma–Waste Interaction and Mixing in a Drift and Conduit

In DOE's TSPA, the amount of waste incorporated into a volcanic conduit is determined by the area of a drift intersected by a stylized cylindrical conduit. This model assumes that waste from disrupted packages located outside the boundary of the conduit will not be entrained into the upward-flowing magma in the conduit. Additional DOE analyses (SAR Section 2.3.11.3.4.4) described how circulation of magma and gas might occur between a conduit and other parts of the intersected drift. However, DOE did not characterize the extent or magnitude of this circulation or evaluate the potential for this circulation to entrain small particles of degraded waste from elsewhere in a drift beyond the conduit. Additional degraded waste may be available, as DOE assumed that the waste form is instantly degraded when the waste package fails during the intrusive event (SAR Section 2.3.11.3.2.4).

The NRC staff reviewed the DOE information in SAR Section 2.3.11.3.4.4 and associated published literature. NRC staff notes that although DOE did not evaluate the effects of potential magma circulation, the significance of these potential effects would be less than effects associated with secondary conduit development. As Menand, et al. (2008aa) discussed, magma circulation in an intersected drift has the potential to transport some sizes of waste particles into the erupting conduit, if flow conditions in the drift are appropriate. The NRC staff expects that only a relatively small amount of waste particles could potentially be transported by magma circulation, because materials on the floor of the drift (e.g., pallets on which the waste canisters rest, damaged/degraded engineered barrier system materials, and the invert; SAR Section 2.1) would present obstacles to magma flow (SNL, 2007ag,ar; Detournay, et al., 2003aa; Darteville and Valentine, 2009aa). These obstacles would also present rough surfaces that would impede waste particles from entrainment in the circulating magma and thus limit the amount of waste that could be released in an eruption. Therefore, the potential increase in entrained waste due to magma circulation would be significantly less than an order of magnitude (i.e., much less than the amount of waste contained in a potentially intersected drift). By providing an analysis showing that the expected annual dose increases linearly with increasing source term for the volcanic modeling case, as detailed in SAR Section 2.3.11.3.2.2 and SNL Appendix P (2008ag), DOE has provided information that shows the potential effects of magma circulation are not significant to dose estimates from the igneous scenario.

Further, the NRC staff reviewed the information DOE provided in SAR Section 2.3.11.3.2.2 to evaluate the potential effects of secondary conduits developing away from the location of dike intersection with the drift. The NRC staff notes that although the likelihood of secondary conduits relies on a series of unusual conditions and thus appears remote, DOE has not provided a technical basis to determine this likelihood. However, the NRC staff notes that the DOE assumption that development of a secondary conduit could potentially lead to the eruption of all waste in an intersected drift is reasonable because the assumption is consistent with available information (Woods, et al., 2002aa; Menand, et al., 2008aa; Lejeune, et al., 2009aa; Darteville and Valentine, 2009aa). DOE accounted for the uncertainty associated with secondary conduit formation by providing an analysis in SAR Section 2.3.11.3.2.2 that shows a hypothesized two orders-of-magnitude increase in amount of waste entrained in an eruption might lead to a two orders-of-magnitude increase in expected annual dose for the volcanic modeling case [see also SNL Appendix P (2008ag)]. Thus, DOE has addressed the significance of secondary conduit formation, because the likelihood of secondary conduits appears remote, and the significance to performance appears to be much smaller than the two orders of magnitude presented in SAR Section 2.3.11.3.2.2 (i.e., much less than 10 percent of the total igneous scenario).

In the DOE volcanic eruption modeling scenario, the number of waste packages intersected becomes input in TSPA for calculating the amount of waste erupted, along with the probability that a conduit will develop in a drift containing waste packages. DOE used a Monte Carlo technique to account for parameter uncertainties such as the future time at which an eruption might occur and the possibility that more than one eruption could happen in the future of the repository. DOE calculates a magma partitioning factor (SAR Section 2.3.11.4.2.2.2; SNL, 2007ab) to determine the amount of the waste partitioned into a potential volcanic tephra fall deposit, the only volcanic product that is significant to dose (TER Section 2.2.1.3.13.3.1). DOE determined that 10 to 50 percent of the total amount of waste entrained in an eruption will be in the resulting tephra fall deposit. The magma partitioning factor and the expected style of eruption (violent Strombolian) from the volcanic conduit(s) is evaluated as part of the abstraction for airborne volcanic transport in TER Section 2.2.1.3.13.

DOE proposed that the amount of waste particles incorporated into the erupting magma would only constitute a minor amount (trace phase) in the magma in all DOE's scenarios and that its presence would not be expected to influence the eruptive behavior of the magma (SNL, 2007ab). The NRC staff notes that DOE's estimate of the amount of the waste that could become incorporated into the fallout deposit is documented and supported. Independent NRC staff calculations substantiate DOE's claim that the amount of waste transported in a conduit and into the tephra deposit would be on the order of 10^{-6} of the concentration of tephra at any point in the deposit (SAR Section 2.3.11.4.2.2.3). NRC staff notes that the waste particles will not affect the eruptive processes occurring in the magma (SNL, 2007ab) and that the style of eruption would not be influenced by the presence of the waste. Therefore, it is reasonable for DOE to model dispersal and fall for airborne transport of radionuclide-contaminated tephra (reviewed in TER Section 2.2.1.3.13) on the basis of past and current similar style volcanic activity.

Summary of the NRC Staff's Review of the Volcanic Eruption Modeling Case

DOE has provided information to support the modeling approach used to represent volcanic igneous events in the performance assessment. DOE has represented how potential volcanic conduits could form randomly along an igneous intrusion and entrain waste. The NRC staff notes that DOE assumed that all waste packages located within the footprint of a potential drift-intersecting conduit would release all degraded waste into the volcanic eruption. DOE has accounted for uncertainties in the amount of waste that potentially could be disrupted and erupted during a modeled volcanic event by providing calculations of dose sensitivity to amount of waste erupted (see TER Section 2.2.1.4.1.3.3.2). These calculations form the basis for NRC staff understanding that uncertainties associated with the potential effects of magma circulation in a drift, or the remote chance of secondary conduit development, would not affect dose significantly. DOE has provided a basis for the use of the magma partitioning factor and has supported that basis with information from suitable volcanic analogs. The NRC staff notes that

DOE has represented the potentially significant effects of subsurface volcanic processes and events in the performance assessment.

The NRC staff notes that the incorporation factor of waste into magma adopted by DOE (the entire contents of waste packages are assumed to be mixed into the magma) is conservative as it allows for greater amounts of incorporated waste than might realistically occur. There are inherent uncertainties in how much waste would be incorporated into erupting magma because no suitable natural analogs have been identified for this process. Potential waste incorporation is highly dependent on the behavior of the magma during the interaction, particularly on the extent of fragmentation, which is driven by degassing of the partially solidified magma and depends on many variables. Magma fragmentation may occur as deep as repository depths {300 m [~1,000 ft]} in violent Strombolian eruptions (Doubik and Hill, 1999aa). However, in most Strombolian eruptions, fragmentation is modeled to occur via explosive gas bubbles bursting at less than 100 m [328 ft] below the surface (e.g., Wilson and Head, 1981aa). The presence of repository drifts could cause deeper fragmentation than in typical volcanic environments (e.g., Woods, et al., 2002aa), and DOE's model for waste incorporation relies on a vigorously degassing, partly fragmenting magmatic environment in the drifts. Despite these uncertainties, the NRC staff notes that these factors will not make a significant difference to the dose from the volcanic eruption modeling scenario as calculated in the DOE TSPA, because the likely increase in the amount of waste incorporated into the erupted magma is within the range of uncertainty considered.

2.2.1.3.10.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for igneous disruption of waste packages is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.10.5 References

BSC. 2005af. "Magma Dynamics at Yucca Mountain, Nevada." ANL–MGR–GS–000005. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2003ab. "Technical Basis Document No. 13: Volcanic Events." Rev. 02. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

CRWMS M&O. 1996aa. "Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada." BA0000000–01717–2200–00082. Rev. 0. Las Vegas, Nevada: CRWMS M&O.

Dartevelle, S. and G.A. Valentine. 2009aa. "Multiphase Magmatic Flows at Yucca Mountain, Nevada." *Journal of Geophysical Research, Solid Earth*. Vol. 113. pp. B12209. doi:10.1029/2007JB005367.

Dartevelle, S. and G.A. Valentine. 2005aa. "Early Time Multiphase Interactions Between Basaltic Magma and Underground Repository Openings at the Proposed Yucca Mountain Radioactive Waste Repository." *Geophysical Research Letters*. Vol. 32. pp. L22311. doi:1029/2005GL024172.

Delaney, P.T. and A.E. Gartner. 1997aa. "Physical Processes of Shallow Mafic Dike Emplacement Near the San Rafael Swell, Utah." *Geological Society of America Bulletin*. Vol. 109, No. 9. pp. 1,177–1,192.

Detournay, E., L.G. Mastin, J.R.A. Pearson, A.M. Rubin, and F.J. Spera. 2003aa. "Final Report of the Igneous Consequences Peer Review Panel." DN2000219072. MOL20031014:0097. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

DOE. 2009av. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

Doubik, P. and B.E. Hill. 1999aa. "Magmatic and Hydromagmatic Conduit Development During the 1975 Tolbachik Eruption, Kamchatka, With Implications for Hazards Assessment at Yucca Mountain, NV." *Journal of Volcanology and Geothermal Research*. Vol. 91. pp. 43–64.

Hill, B.E. and C.B. Connor. 2000aa. "Technical Basis for Resolution of the Igneous Activity Key Technical Issue." ML011930254 San Antonio, Texas: CNWRA.

Keating, D.N., G.A. Valentine, D.J. Krier, and F.V. Perry. 2008aa. "Shallow Plumbing Systems for Small-Volume Basaltic Volcanoes." *Bulletin of Volcanology*. Vol. 70. pp. 563–582.

Lejeune, A., B.E. Hill, A.W. Woods, R.S.J. Sparks, and C.B. Connor. 2009aa. "Intrusion Dynamics for Volatile-Poor Basaltic Magma Into Subsurface Nuclear Installations." *Volcanic and Tectonic Hazard Assessment for Nuclear Facilities*. C.B. Connor, N.A. Chapman, and L.J. Connor, eds. Cambridge, United Kingdom: Cambridge University Press.

Menand, T., J.C. Phillips, and R.S.J. Sparks. 2008aa. "Circulation of Bubbly Magma and Gas Segregation Within Tunnels of the Potential Yucca Mountain Repository." *Bulletin of Volcanology*. Vol. 70. pp. 947–960.

Nicholis, M.G. and M.J. Rutherford. 2004aa. "Experimental Constraints on Magma Ascent Rate for the Crater Flat Volcanic Zone Hawaiite." *Geology*. Vol. 32. pp. 489–492.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

Pioli, L., E. Erlund, E. Johnson, K. Cashman, P. Wallace, M. Rosi, and H. Delgado Granados. 2008aa. "Explosive Dynamics of Violent Strombolian Eruptions: The Eruption of Paricutin Volcano 1943–1952 (Mexico)." *Earth and Planetary Science Letters*. Vol. 271. pp. 359–368.

SNL. 2008ac. "Features, Events, and Processes for the Total System Performance Assessment: Methods." ANL–WIS–MD–000026. Rev. 00. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ab. "Atmospheric Dispersal and Deposition of Tephra From a Potential Volcanic Eruption at Yucca Mountain, Nevada." MDL–MGR–GS–000002. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ae. "Characterize Eruptive Processes at Yucca Mountain, Nevada." ANL–MGR–GS–000002. Rev. 03. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ag. "Dike/Drift Interactions." MDL–MGR–GS–000005. Rev. 02. ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ar. "Number of Waste Packages Hit by Igneous Events." ANL–MGR–GS–000003. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2005ae. "In-Package Chemistry Abstraction." ANL–EBS–MD–000037. Rev. 04. ADD 01. Las Vegas, Nevada: Sandia National Laboratories.

Thorarinsson, S., S. Steinthorsson, T. Einarsson, K. Kristmannsdottir, and N. Oskarsson. 1973aa. "The Eruption on Heimaey, Iceland." *Nature*. Vol. 241. pp. 372–375.

Valentine, G.A. and K.E.C. Krogh. 2006aa. "Emplacement of Shallowing Dikes and Sills Beneath a Small Basaltic Volcanic Center—The Role of Pre-Existing Structure (Paiute Ridge, Southern Nevada, USA)." *Earth and Planetary Science Letters.* Vol. 246, No. 3. pp. 217–230.

Valentine, G.A. and K.R. Groves. 1996aa "Entrainment of Country Rock During Basaltic Eruptions of the Lucero Volcanic Field, New Mexico." *Journal of Volcanology and Geothermal Research*. Vol. 161, No. 1-2. pp. 57–80.

Valentine, G.A., D.J. Krier, F.V. Perry, and G. Heiken. 2007aa. "Eruptive and Geomorphic Processes at the Lathrop Wells Scoria Cone Volcano." *Journal of Volcanology and Geothermal Research.* Vol. 161, No. 1–2. pp. 57–80.

Valentine, G.A., F.V. Perry, D.J. Krier, D.N. Keating, R.E. Kelley, and A.H. Cogbill. 2006aa. "Small-Volume Basaltic Volcanoes: Eruption Products and Processes, and Posteruptive Geomorphic Evolution in Crater Flat (Pleistocene), Southern Nevada." *Geological Society of American Bulletin*. Vol. 118, No. 5. pp. 1,313–1,330.

Wilson L. and J.W. Head, III. 1981aa. "Ascent and Eruption of Basaltic Magma on the Earth and Moon." *Journal of Geophysical Research*. Vol. 86, No. B4. pp. 2,971–3,001.

Woods, A.W., S. Sparks, O. Bokhove, A. Lejeune, C.B. Connor, and B.E. Hill. 2002aa. "Modeling Magma-Drift Interaction at the Proposed High-Level Radioactive Waste Repository at Yucca Mountain, Nevada, USA." *Geophysical Research Letters*. Vol. 29, No. 13. p. 1,641.

(Intentionally Left Blank)

CHAPTER 14

2.2.1.3.12 Concentration of Radionuclides in Groundwater

2.2.1.3.12.1 Introduction

This section of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff review of information DOE provided in the Safety Analysis Report (SAR) (DOE, 2008ab) on the concentration of radionuclides in groundwater extracted by pumping and used in the annual water demand. The NRC staff reviewed the methods and assumptions DOE used to estimate groundwater radionuclide concentrations. The NRC staff review focused on SAR Sections 2.3.9 and 2.4.4. SAR Section 2.3.9 includes discussions of saturated zone radionuclide transport and groundwater. SAR Section 2.4.4 includes the analysis of repository performance with respect to the protection of groundwater.

2.2.1.3.12.2 Evaluation Criteria

The NRC staff's review of the DOE's calculation of the concentration of radionuclides in the groundwater is guided by 10 CFR 63.312(c), which states that the reasonably maximally exposed individual (RMEI) uses well water with average concentrations of radionuclides based on an annual water demand of 3.7×10^9 L [3,000 acre-ft]. The NRC staff followed a risk informed approach and the guidance in the Yucca Mountain Review Plan (YMRP; NRC, 2003aa) to evaluate DOE calculation/analysis of the concentration of radionuclides in groundwater.

2.2.1.3.12.3 Assessment of Well Water Concentration Estimates

In SAR Section 2.4.4, DOE stated that it explicitly assumed that all radionuclides transported by groundwater from the Yucca Mountain disposal system in a given year are captured in the annual water demand of 3.7×10^9 L [3,000 acre-ft]. DOE determined the annual mean concentrations of transported radionuclides in the saturated-zone groundwater by dividing the annual mass flux of radionuclides reaching the accessible environment boundary by the annual water demand (SAR Section 2.4.4.1.1.1). As DOE presented in SAR Section 2.3.9, this annual mass flux includes both those radionuclides explicitly transported in the TSPA model and those calculated assuming secular equilibrium in long-lived decay chains.

NRC Staff Review

YMRP Section 2.2.1.3.12 states that if DOE assumes that all radionuclides that reach the reasonably maximally exposed individual in a given year are included in the pumping wells with annual water demand of 3.7×10^9 L [3,000 acre-ft], then a simplified review should focus on the bounding assumptions. In SAR Section 2.4.4, DOE stated that it explicitly assumed that all radionuclides transported by groundwater from the Yucca Mountain disposal system in a given year are captured in the annual water demand of 3.7×10^9 L [3,000 acre-ft]. Thus, the NRC staff followed the simplified review approach in YMRP Section 2.2.1.3.12. The NRC staff verified that DOE determined the annual mean concentrations of transported radionuclides in the saturated-zone groundwater by dividing the annual mass fluxes of radionuclides reaching the location of the RMEI by the annual water demand (SAR Section 2.4.4.1.1.1). The NRC staff evaluation of the radionuclide mass flux is provided in TER Section 2.2.1.3.9. DOE's saturated zone transport abstraction model (SAR Section 2.3.9) explicitly tracks transport of a set of contaminant radionuclides, and assumes that decay-chain daughter radionuclides are in secular equilibrium with their parents at the location of the RMEI. As discussed in TER Section 2.2.1.3.9, this assumption may not be reasonable for cases where a long-lived parent radionuclide is more strongly sorbed than its decay products. In its response to the NRC staff's request for additional information, DOE evaluated this effect and showed that for the conditions expected in the saturated-zone transport path, the magnitude of the predicted excess daughter activity is not significant for performance (DOE, 2009de). The NRC staff notes in TER Section 2.2.1.3.9 that, including the uncertainty from possible excess activity of decay-chain daughter radionuclides, DOE's representation of the annual mass fluxes of radionuclides reaching the location of the RMEI is reasonable. The NRC staff notes that the DOE's calculation is consistent with the YMRP guidance because DOE showed that the reasonably maximally exposed individual uses well water with an average concentration of radionuclides based on an annual water demand of 3.7×10^9 L [3,000 acre-ft].

2.2.1.3.12.4 NRC Staff Conclusions

The NRC staff notes that the DOE's description of the calculation of the concentration of radionuclides in groundwater is consistent with the guidance in the YMRP. The NRC staff notes that the technical approach is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.12.5 References

DOE. 2009de. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.9), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.9, Set 1." Letter (November 16) J.R. Williams to J.H. Sulima (NRC). ML092820675. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 0. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

CHAPTER 15

2.2.1.3.13 Airborne Transport and Redistribution of Radionuclides

2.2.1.3.13.1 Introduction

This chapter evaluates the U.S. Department of Energy's (DOE) information on airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption following igneous disruption of waste packages. It also evaluates DOE information on the redistribution of those radionuclides in soil. This evaluation of DOE's performance assessment for the volcanic eruption modeling case is a sequel to the evaluation of possible igneous disruption of the proposed repository [DOE's igneous intrusion modeling case; see Technical Evaluation Report (TER) Section 2.2.1.3.10]. This chapter also evaluates redistribution of radionuclides in soil in the accessible environment, which in DOE's model arrives in the accessible environment via groundwater transport. The U.S. Nuclear Regulatory Commission (NRC) staff's evaluation is based on information in the DOE Safety Analysis Report (SAR) (DOE, 2009av), as supplemented by DOE's responses (DOE, 2009bk–bm) to the NRC staff's requests for additional information (RAIs).

This chapter addresses 2 of the 14 model abstraction sections indicated in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa): airborne transport of radionuclides (YMRP Section 2.2.1.3.11) and redistribution of radionuclides in soil (YMRP Section 2.2.1.3.13). The NRC staff's assessment of information in DOE's SAR for these two abstraction sections used the guidance in the YMRP to conduct a risk-informed review. Together, airborne transport of radionuclides during a potential future explosive volcanic eruption that generates tephra (ash) and redistribution of radionuclides deposited on the landscape by that eruption constitute DOE's volcanic ash exposure scenario in its biosphere model for the Total System Performance Assessment (TSPA) (SAR Section 2.3.10.2.6). As part of the review of redistribution of radionuclides, the NRC staff evaluated DOE's performance assessment for the exposure scenario where radionuclide-contaminated groundwater may cause the reasonably maximally exposed individual (RMEI) to be exposed to a dose (SAR Section 2.3.10.2.3). SAR Figure 2.3.10-1 displayed a separate flow of information for the volcanic ash exposure scenario compared to the groundwater exposure scenarios in the DOE performance assessment. This chapter reflects this separation of information and presents the NRC staff's review and evaluation, first for the volcanic ash exposure scenario and second for the groundwater exposure scenario.

For the volcanic ash exposure scenario, the NRC staff evaluated the following three abstracted models addressed in DOE's SAR:

1. Airborne transport, dispersion, and deposition of tephra and high-level waste

2. Redistribution by fluvial (running water or stream) transport of contaminated tephra within the Fortymile Wash catchment basin, mixing and dilution with noncontaminated sediment, and deposition of the tephra-sediment mixture on the Fortymile Wash alluvial fan at the reasonably maximally exposed individual location

3. The downward migration of radionuclides in the soil at the alluvial fan in the accessible environment

The latter two abstracted models comprise DOE's performance assessment for redistribution of radionuclides in soil (YMRP Section 2.2.1.3.13), while the first abstracted model constitutes DOE's performance assessment for airborne transport of radionuclides (YMRP Section 2.2.1.3.11).

For the groundwater exposure scenario, this chapter presents the NRC staff evaluation of DOE's surface soil submodel, which is also part of the performance assessment for redistribution of radionuclides in soil. For both exposure scenarios, the final outputs of the abstractions evaluated in this chapter are radionuclide concentrations in soil, which are direct inputs to the DOE biosphere model for calculating annual doses to the reasonably maximally exposed individual (reviewed by the staff in TER Section 2.2.1.3.14). Associated with this, TER Section 2.2.1.3.4 presents the NRC staff evaluation of the radionuclide inventory, which is an input to the volcanic ash exposure scenario (SAR Figure 2.3.10-3). TER Section 2.2.1.4 provides the NRC staff's evaluation of the overall TSPA.

2.2.1.3.13.2 Evaluation Criteria

The NRC staff's review of model abstractions used in DOE's postclosure performance assessment, including those considered in this chapter for airborne transport and redistribution of radionuclides, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The DOE Total System Performance Assessment model is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of features, events, and processes is presented in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, DOE should

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those features, events, and processes used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction of airborne transport and redistribution of radionuclides involves igneous activity. 10 CFR 63.342(c)(1) provides requirements for assessing the effects of seismic and igneous activity on the repository performance, subject to the probability limits given in 10 CFR 63.342(a) and (b). Specific constraints on seismic and igneous activity analyses are in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

In addition,10 CFR 63.305 states the following requirements for characteristics of the reference biosphere, as used in this abstraction for redistribution of radionuclides in soil:

- Features, events, and processes that describe the reference biosphere must be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site [10 CFR 63.305(a)].

- DOE should not project changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology; all analyses done to calculate dose must assume that all of those factors are constant as they are at the present [10 CFR 63.305(b)].

- DOE must vary factors related to the geology, hydrology, and climate based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, consistent with the requirements for performance assessments specified at 10 CFR 63.342.

- Biosphere pathways must be consistent with arid or semi-arid conditions [10 CFR 63.305(d)].

The NRC staff review of the SAR and supporting information follows the guidance laid out in YMRP Sections 2.2.1.3.11, Airborne Transport of Radionuclides, and 2.2.1.3.13, Redistribution of Radionuclides in Soil, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria that provide guidance for the NRC staff's evaluation of DOE's model abstraction of airborne transport and redistribution of radionuclides are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

NRC staff review used a risk-informed approach and the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of airborne transport and redistribution of radionuclides important to repository performance. The NRC staff considered all five criteria provided in the Yucca Mountain Review Plan in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as assessed by the NRC staff, are discussed in detail in this chapter. The NRC staff's assessment is based both on risk information provided by DOE, and on NRC staff knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.13.3 Technical Evaluation

In SAR Figure 2.3.11-1, DOE presented the information flow for the volcanic eruption modeling case. As stated previously, DOE's abstracted model on atmospheric dispersal and deposition of tephra constitutes its performance assessment of airborne transport of radionuclides. DOE's abstracted models for tephra redistribution and vertical radionuclide migration in soil together comprise the performance assessment of redistribution of radionuclides in soil.

Airborne transport of radionuclides pertains to the volcanic ash exposure scenario, which involves a possible disruption of the Yucca Mountain repository by a future volcanic eruption. In this scenario, high-level radioactive waste is mixed with magma and ejected into the atmosphere incorporated within the volcanic tephra {fragments of cooled magma that are transported through the air, including ash particles that have diameters less than 2 mm [0.08 in]}. The airborne transport abstracted model accepts the number of waste packages intersected by volcanic conduits, provided in SAR Section 2.3.11.4.2.1 and evaluated in TER Section 2.2.1.3.10, and estimates the concentration and thickness of radionuclide-contaminated tephra that could be deposited on the ground surface of the Yucca Mountain region (SAR Figure 2.3.11-1). As depicted in SAR Figure 2.3.11-1, DOE then uses this information as input to the volcanic ash exposure scenario (SAR Section 2.3.10) for estimating the dose to the reasonably maximally exposed individual via surface redistribution of contaminated tephra and by migration of radionuclides from tephra particles into the soil, as described next.

Redistribution abstracted models together calculate the time-dependent profile of radionuclide concentration in the contaminated soil horizon at the reasonably maximally exposed individual location. The DOE airborne transport abstracted model (described previously) provides input on the tephra deposit for the tephra redistribution calculations of waste concentrations in redistributed tephra. Another DOE redistribution-related abstracted model uses this information to estimate the downward migration of radionuclides from tephra into soil at the alluvial fan of Fortymile Wash and calculate the concentration of waste in redistributed tephra at the reasonably maximally exposed individual location (SAR Figure 2.3.11-1). Waste concentration information from DOE's redistribution models is coupled with information on the radionuclide inventory (radionuclide activities per unit mass of waste) to yield radionuclide concentration profiles in soil. The fraction of tephra that can be resuspended and inhaled by the reasonably maximally exposed individual during activities such as tillage is also important; this is the dominant exposure pathway for the first 10,000 years after repository closure in DOE's performance assessment analysis.

In this section, the NRC staff also evaluates the DOE surface soil submodel for the groundwater exposure scenario, described in SAR Section 2.3.10. In this model, radionuclides are considered to be added to the surface soil from irrigation with contaminated groundwater. The surface soil submodel accepts the concentration of radionuclides in groundwater in the accessible environment (as provided in SAR Section 2.4.4 and reviewed in TER Section 2.2.1.3.12) and calculates loss of radionuclides from the surface soil via mechanisms such as radioactive decay, leaching into deeper zones, erosion of soil particles, and gaseous releases to the atmosphere. As depicted in SAR Figures 2.3.10-1 and 2.3.10-10, the output from the surface soil model is used by the rest of DOE's biosphere model, which is reviewed in TER Section 2.2.1.3.14.

NRC Staff Perspective on Risk

The volcanic ash exposure scenario and groundwater exposure scenario provide different contributions to repository performance in the DOE assessment. The NRC staff has evaluated DOE's assessment that the volcanic ash exposure scenario (volcanic eruption modeling case) does not significantly influence repository performance. DOE shows that its mean dose contribution is more than a factor of 1,000 smaller than the overall peak dose within the initial 10,000 years and more than a factor of 10,000 smaller than the overall peak dose after 10,000 years (SAR Figure 2.4-18). Further, DOE's dose exposure assessment is consistent with the NRC staff's independent analyses (see TER Section 2.2.1.4.1.3.3.2). The remaining DOE modeling cases depicted in SAR Figure 2.4-18 constitute the groundwater exposure scenario. The groundwater exposure scenario dominates the overall peak dose within 10,000 years and after 10,000 years (SAR Figure 2.4-18). Although this risk information suggests that the NRC staff should focus on the surface soil submodel in the groundwater exposure scenario and only conduct a simplified review of the volcanic ash exposure focusing on the bounding assumptions, the NRC staff also used DOE's multiple barrier information, consistent with YMRP Section 2.2.1.3, to inform its review.

The NRC staff's review of SAR Section 2.1.3 determined that no aspect of the material reviewed in this chapter is identified as a barrier. TER Section 2.2.1.1 describes the NRC staff's evaluation of multiple barriers. However, DOE did identify that a volcanic event could adversely affect the engineered barrier system's ability to prevent the release or reduce the release rate of radionuclides from the waste, and to prevent or reduce the movement of radionuclides away from the repository (SAR Section 2.3.11.1) by destroying the waste packages and releasing the contained radionuclides in the erupting material (SAR Section 2.1.2.2.5). On the basis of the NRC staff's review in TER Section 2.2.1.3.13.3.1, the NRC staff notes that DOE's technical basis for barrier capability was based on, and is consistent with, the technical basis for the performance assessment.

For the volcanic ash exposure scenario, the NRC staff's evaluation of DOE's performance assessment is presented in TER Section 2.2.1.4.1.3.2. The four input quantities identified for that evaluation (fraction entrained in ash, tephra volume, tephra density, and ash areal concentration) directly relate to the airborne transport abstracted model reviewed in this chapter. All three abstracted models evaluated in this chapter for the volcanic ash exposure scenario account for the bulk transport of radionuclides in waste and do not include processes that separate radionuclides or transport radionuclides at different rates. For this reason, the evaluation of the abstracted models for volcanic ash exposure in this chapter does not focus on individual radionuclide contributions to total dose, which are considered in TER Section 2.2.1.4.1.3.3.2.

However, an understanding of the dominant radionuclides, and their exposure pathways, that contribute to dose provides risk insights into the important aspects of the performance assessment for the DOE volcanic eruption modeling case. SAR Figure 2.4-32 identified the contribution of radionuclides to mean annual dose for the volcanic eruption modeling case. SAR Table 2.3.10-15 identified the average percentage exposure pathway contributions to the annual dose for the volcanic ash exposure scenario. On the basis of its review of DOE's information, NRC staff notes that at early times (i.e., before 500 years), the overall dose is dominated by six radionuclides: Sr-90, Cs-137, Pu-238, Pu-239, Pu-240, and Am-241. At longer times (i.e., after 5,000 years), the dose is dominated by Pu-239 and Pu-240, and at very long times (i.e., after 100,000 years), the dose is dominated by Ra-226 (SAR Figure 2.4-32). On the basis of its identification of the dominant dose contributors and review of information in DOE's SAR Table 2.3.10-15, NRC staff notes that inhalation of particulates from the resuspension of contaminated tephra deposits is the dominant dose pathway for 10,000 years. After about 100,000 years, for Ra-226, the dominant exposure pathway is external exposure (SAR Table 2.3.10-15). The NRC staff's independent analyses, documented in NRC, Volume 2, Appendix D (2005aa), gave similar results about the dominant exposure pathway. Thus, the NRC staff has reviewed the performance assessment to (i) focus on those processes that most affect the concentration of waste in the resuspendable layer and (ii) focus on those processes that most affect the concentration of waste in the soil layers that control external exposure.

For the groundwater exposure scenario, the surface soil submodel reviewed in this chapter is one component of the abstracted model for the biosphere that calculates biosphere dose conversion factors. For the radionuclides Tc-99, I-129, Np-237, and Pu-242 discussed in TER Section 2.2.1.4.1.3 in the table on groundwater biosphere dose conversion factors and TER Section 2.2.1.4.1.3.3, the pathways linked to the surface soil submodel account for up to 50 percent of the radionuclide biosphere dose conversion factor, as identified in SNL Tables 6.13-1 and 6.13-2 (2007ac).

On the basis of these risk considerations, the NRC staff conducted a risk-informed review of airborne transport of radionuclides and redistribution of radionuclides in the soil. The NRC staff focused on those aspects of these model abstractions that impact calculated dose to the reasonable maximally exposed individual. To assess the effect that the combined uncertainties could have on calculated dose, the NRC staff also focused on those aspects that could cause at least a factor of two effect on intermediate model outputs over the range of an individual parameter value.

As identified previously, SAR Figure 2.3.10-1 displayed a separate flow of information for the volcanic ash exposure scenario compared to the groundwater exposure scenarios in the DOE performance assessment. The NRC staff's evaluations of the DOE information on the volcanic ash exposure scenario and the surface soil submodel for the groundwater exposure scenario are documented in TER Sections 2.2.1.3.13.3.1 and 2.2.1.3.13.2, respectively.

2.2.1.3.13.3.1 Assessment and Review of the Volcanic Ash Exposure Scenario

The NRC staff's evaluations of DOE's abstracted models on (i) airborne transport, dispersion, and deposition of tephra and high-level waste, (ii) redistribution of tephra, and (iii) the vertical movement of radionuclides in the soil at the alluvial fan in the accessible environment are presented separately in three subsections.

In addition to reviewing the individual abstracted models, the NRC staff also reviewed how DOE implemented these models into the Total System Performance Assessment (TSPA). To place the individual model abstractions into the framework of the TSPA analysis, the NRC staff summarizes DOE's implementation of the volcanic eruption modeling case next.

DOE integrated abstracted models of TSPA analysis for the volcanic eruption modeling case (volcanic interaction with the repository, atmospheric transport, tephra redistribution, volcanic ash exposure) in a GoldSim modeling environment. DOE used the initial radionuclide inventory from a "blended" waste package to calculate radionuclide transport, as described in the review of the previously mentioned submodels. A blended waste package inventory was calculated by using a weighted average of commercial spent nuclear fuel and codisposal waste packages and inventories. Following a conditional future eruptive event, tephra transport and redistribution are abstracted to occur instantaneously (i.e., radionuclide waste transport to the reasonably maximally exposed individual is instantaneous) (SAR Section 2.3.11.4.2.3.1). The time dependence of radionuclide diffusion (downward migration) into the soil at the reasonably maximally exposed individual location was accounted for in the tephra redistribution model. The radionuclide concentration in the soil at the reasonably maximally exposed individual location, in g/cm^2, was modified by a "decay factor" to account for radionuclide decay and ingrowth. The resultant source term was provided to the volcanic ash exposure submodel to calculate dose.

2.2.1.3.13.3.1.1 Airborne Transport Modeling

The NRC staff conducted a risk-informed review of airborne transport of radionuclides, concentrating on aspects important to the volcanic ash exposure scenario in the DOE performance assessment.

Important Aspects of Airborne Transport

The abstracted model for atmospheric transport of radionuclides determines the characteristics of contaminated tephra deposited on the surrounding landscape. DOE's analysis results indicated that the following parameters for airborne transport were influential to the volcanic ash exposure scenario: magma partitioning factor, tephra volume, eruptive power and duration, and mean ash particle diameter, wind direction, and wind speed. The magma partitioning factor is a fraction between zero and one and acts as a direct multiplier on the eruption source term and eruptive dose, similar to the number of waste packages entrained into the erupting magma that pertains to the review in TER Section 2.2.1.3.10. DOE's analysis results [e.g., SNL Figures C–1 and C–2 (2007ab)] showed that waste concentration in tephra is sensitive to tephra volume, eruptive power, and mean ash particle diameter. DOE's sensitivity analyses concluded that the initial tephra thickness at the reasonably maximally exposed individual location (near the Fortymile Wash alluvial fan apex) is strongly dependent on wind direction, wind speed, and mean ash particle diameter, and moderately dependent on eruptive power and eruptive duration, as identified in SNL Appendix C (2007ab). DOE found that other parameters of its airborne transport abstracted model were less influential on tephra thickness.

Summary of Information on Airborne Transport

The volcanic eruption modeling case was described in SAR Section 2.3.11.4. In SAR Table 2.3.11-1, DOE identified the features, events, and processes included in the TSPA.

For a repository-drift-penetrating basaltic eruption, DOE modeled the contamination of tephra with waste and the amount of radionuclides contained within the tephra-fall deposit. On the basis of studies of analog volcanoes, DOE apportioned contaminated magma into three volcanic products; namely, lava, scoria cone-forming deposits (selectively composed of the largest tephra fragments), and more widespread tephra-fall deposits. To account for waste that is incorporated in volcanic ejecta that form scoria cones and lava flows, DOE applied a magma partitioning factor for the fraction of waste incorporated with tephra to the total waste erupted. In the DOE model for the extrusive event, only waste incorporated with tephra contributes radiological dose to the reasonably maximally exposed individual; waste apportioned into lava flows and scoria cones does not contribute to dose. The amount of waste incorporated in tephra scales with the magma partitioning factor.

In its igneous eruption modeling, DOE identified that all the explosive phases of the most likely future eruption, on the basis of the interpreted behavior of the youngest volcano near the repository site (Lathrop Wells), are considered to be violent Strombolian (SAR Section 2.3.11.4.1). The eruption would produce plumes of tephra in the atmosphere that could transport particulates, including high-level radioactive waste, downwind from the vent. This process could deposit radionuclides at the reasonably maximally exposed individual location, either from direct sedimentation of contaminated ash particles from the volcanic plume or from the remobilization by wind or surface water of the radionuclide-contaminated volcanic ash after initial deposition. DOE's approach to determining waste concentration in the tephra is sensitive to the tephra volume. For example, smaller tephra volumes result in higher waste concentration in tephra (i.e., waste mass per unit mass of tephra) for the same number of waste packages entrained. In SNL (2008ag), DOE evaluated exposure to airborne concentrations of radionuclides that are captured in the tephra during the eruption (direct tephra-fall exposure) and found that it did not increase expected annual dose significantly due to the extremely short exposure duration. In the following review, "tephra" refers to airborne magmatic fragments of all sizes, whereas "ash" refers specifically to particles less than 2 mm [0.08 in] in diameter.

A violent Strombolian-type eruption is characterized by the development of a sustained, buoyant plume of hot air and volcanic tephra that commonly rises several kilometers [a few miles] above the volcano. DOE modeled the dispersal processes as turbulent advection diffusion using the Suzuki (1983aa) model. The ASHPLUME conceptual model and the ASHPLUME code (Jarzemba, et al., 1997aa), as used by DOE, implement the Suzuki approach to model the dispersal of tephra on the basis of the diffusion of particles from an eruption column, horizontal diffusion of particles by atmospheric turbulence, horizontal advection by atmospheric circulation, and settling of particles by gravity. ASHPLUME accounts for incorporation and entrainment of waste particles into magma during a potential volcanic eruption through the repository and estimates the concentration (expressed as g/cm^2) and thickness of radionuclide-contaminated tephra deposited on the ground surface. Following a conditional eruptive event, tephra transport is abstracted to occur instantaneously (i.e., radionuclide waste transport to the reasonably maximally exposed individual is instantaneous).

In DOE's approach, wind direction significantly affects tephra dispersal and deposit location. Tephra deposits that might fall at the reasonably maximally exposed individual location are strongly dependent on the presence of northerly winds that would transport the tephra plume to the south from a volcanic vent within the repository area (SAR Figure 2.3.11-13). The tephra deposit at the reasonably maximally exposed individual location becomes negligible for winds without a strong northerly component (north, north-northwest, or north-northeast), as identified in SNL Figure C–7 and Table D–5 (2007ab). The majority of the wind vectors at the site result in tephra being deposited to the east of Yucca Mountain (SAR Figure 2.3.11-15). According to SNL Appendix K (2007ab), this wind direction provides a source of material for remobilization within the Fortymile Wash catchment basin. DOE's sensitivity analyses indicated that wind direction produced a greater contribution to dose than plume spread and divergence, as outlined in SNL Figure K–4c (2007ab). These sensitivity analyses also demonstrated that increasing the wind speed causes the tephra deposit center mass to shift downwind.

NRC Staff’s Evaluation of Airborne Transport

The NRC staff reviewed SAR Section 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff’s request for additional information (DOE, 2009bk–bm), the supporting DOE information on atmospheric transport of contaminated tephra presented in SNL (2007ab), and information published in peer-reviewed literature (e.g., Suzuki, 1983aa; Hurst and Turner, 1999aa; Andronico, et al., 2008aa).

Model Integration

Potentially relevant features, events, and processes in DOE’s TSPA model were listed in SAR Table 2.2-1. Model abstractions comprise features, events, and processes that have been screened in from DOE’s scenario analysis. TER Section 2.2.1.2.1 documents the NRC staff’s evaluation of the DOE scenario analysis and features, events, and processes screening. As part of the review of the volcanic modeling case, the NRC staff examined DOE’s information on igneous-related features, events, and processes. TER Section 2.2.1.2.1.3.1 considers DOE’s list of features, events, and processes for the volcanic exposure scenario, including airborne transport. TER Section 2.2.1.2.1.3.2 considers DOE’s screening of all features, events, and processes for the volcanic exposure scenario, including airborne transport. DOE excluded a related process from consideration due to low consequence (FEP 1.2.04.07.0B). That process is concerned with leaching of radionuclides from tephra on the surface into the subsurface and into groundwater, whereby radionuclides could be dispersed via the groundwater transport pathway. DOE’s exclusion of this process, identified in DOE Enclosure 1 (2009ab), is reasonable because the possible contribution to mean dose via this mechanism is considerably lower than contributions from the other modeling cases. The NRC staff’s review of the airborne transport abstracted model evaluates DOE’s implementation of the only included process (FEP 1.2.04.07.0A) associated with airborne transport modeling.

FEP 1.2.04.07.0A describes finely divided waste particles that may be erupted from a volcanic vent and deposited on the land surface from a waste-particle-contaminated ash (tephra) cloud or plume. This process is included in the performance assessment through the modeling of an eruption that includes airborne transport and tephra deposition (SNL, 2008ab).

The NRC staff evaluated modeling assumptions and integration in the DOE airborne transport abstracted model. DOE assumed that the tephra in a future eruption would be dispersed by a violent Strombolian eruption column, characterized by heating of entrained air. In its model, the vertical atmospheric transport of the fragmented magma and gas mixture is represented as a thermally buoyant plume. The column rises to an altitude of neutral buoyancy compared to the surrounding atmosphere, at which point it spreads laterally and the resulting plume (ash or tephra cloud) is transported downwind. DOE modeled the dispersal of ash using a peer-reviewed model Suzuki (1983aa) originally developed. Because the scale of horizontal atmospheric turbulence is much greater than the scale of vertical turbulence for violent Strombolian plumes within tens of kilometers [up to 50 mi] of the vent, horizontal diffusion is the dominant factor in the model in determining the width of the plume as it advects downwind. Therefore, the ASHPLUME conceptual model DOE used to simulate tephra deposition is based on a two-dimensional advection-diffusion model in which turbulent diffusion is considered only in the horizontal plane. DOE’s treatment of these processes and assumptions is based on well-established modeling techniques (Sparks, et al., 1997aa).

In its tephra-fall modeling, DOE assumed violent Strombolian activity for the entire duration of the tephra-forming activity. Although violent Strombolian eruptions may have interruptions in activity, modeling such an eruption as a continuous process will not underestimate the amount and character of tephra modeled for an eruption and thus will not underestimate dose to the reasonably maximally exposed individual. In addition, DOE determined the initial plume rise velocity using a relationship among eruptive power, eruption duration, conduit diameter, and eruption column height and plume conditions for violent Strombolian eruptions, which would not underestimate the amount and character of tephra fall deposit (and incorporated waste) modeled for an eruption.

DOE calculated tephra and waste deposited at the reasonably maximally exposed individual location for a point located 18 km [11 mi] south of the volcanic vent, as outlined in SNL Section 6.5.2.1.17 and Table 8-2 (2007ab). DOE's results showed that the point assumption is conservative and does not underestimate the concentration of waste deposited in the 33-km^2 [13-mi^2] reasonably maximally exposed individual location used in the Fortymile Wash Ash Redistribution (FAR) model as part of the Fortymile Wash catchment basin, as identified in DOE Enclosure 9 (2009bk). Results of supplemental ASHPLUME simulations, discussed in DOE Enclosure 9 (2009bk), demonstrated that single point values for tephra and waste measured 18 km [11 mi] south of the potential repository would typically represent the maximum concentrations that would be deposited at the reasonably maximally exposed individual site.

For the particle sizes in the atmospheric transport model of tephra, the model Suzuki (1983aa) developed is appropriate for particles of a mean diameter greater than tens of micrometers [about 6×10^{-4} - 1.2×10^{-3} in]. This cutoff is generally accepted to be the lower limit for the importance of simple gravitational settling of particles because the fall of smaller particles is governed by different physical laws than the Stokes settling assumption of the Suzuki model (Suzuki, 1983aa; Heffter and Stunder, 1993aa). The mass of ash particles smaller than 15 microns [6×10^{-4} in] is less than 2 percent of the total ash mass in most of DOE's model realizations (DOE, 2009bm) and is consistent with violent Strombolian tephra deposits (Andronico, et al., 2008aa).

Data Sufficiency and Data Uncertainty

The NRC staff evaluated data sufficiency and uncertainty in the ASHPLUME model. DOE values for many input parameters in the ASHPLUME model were developed using analogous small-volume basaltic volcanic systems (SNL, 2007ab,ae), which is the commonly used approach when modeling volcanic eruptions (e.g., Hill, et al., 1998aa). Analogous historic violent Strombolian eruptions cited as sources of parameters for use in the ASHPLUME model include Tolbachik, Russia (1975); Parícutin, Mexico (1943–1952); and Cerro Negro, Nicaragua (1850–1999). DOE developed eruption parameter distributions on the basis of empirical relationships from available field data from the deposits of the aforementioned eruptions. The NRC staff reviewed the range of important parameters DOE derived from analog volcanoes (SNL, 2007ab,ae). The NRC staff's evaluation included the parameters for magma partitioning factor, magma volume, ash particle diameter, eruptive power and duration, wind direction and speed, and an eruption column parameter (the diffusion constant, β). These parameters are evaluated individually in the following paragraphs.

Magma Partitioning Factor

DOE accounted for a proportion of disrupted and erupted waste that is partitioned into erosion-resistant products (scoria cone and lava flows) by using a magma partitioning factor with a uniform distribution from 0.1 to 0.5 (also discussed in TER Section 2.2.1.3.10). This range is based on volumetric proportions of cones and lava flows to total erupted volume estimated from field measurements at analog volcanoes, as identified in SNL Section 6.5.2.22 (2007ab). DOE identified in SAR Section 2.3.11.4.1.1.3 that very little erosional modification of lava fields of ~ 350,000-year-old volcanoes (Little Black Peak and Hidden Cone) has occurred. DOE also indicated that little if any cone scoria at the 80,000-year-old Lathrop Wells volcano has yet been remobilized to the base of the cone where it would be available for fluvial transport (SAR Section 2.3.11.4.1.1.3). On the basis of this information provided by DOE, the NRC staff notes the assumption is reasonable that the proportion of magma ending up as cones and flows would not contribute to dose at the reasonably maximally exposed individual location.

DOE used data from eight analog volcanic eruptions to determine a range of 0.1 to 0.5 for the magma partitioning factor (BSC, 2003ad; SNL, 2007ab); estimated doses from the volcanic ash exposure scenario are directly proportional to the magma partitioning factor. However, in the NRC staff's view, not all the analog eruptions cited by DOE showed significant violent Strombolian behavior, and those eruptions that did tended to have magma partitioning factors greater than 0.3. DOE used analog data to support the parameter range for the magma partitioning factor, including the lower part of the range with values less than 0.3, as discussed in DOE Enclosure 8, Table 1 (2009bk). DOE Enclosure 8 (2009bk) stated the tephra component is small in many of these basaltic analog eruptions, which is not typical of violent Strombolian eruptions. The NRC staff notes that eruptions of volcanoes such as Cinder Cone in California, cited as an analog in DOE Enclosure 8 (2009bk), featured only very minor phases of the violent Strombolian activity representative of this style of eruption (Heiken, 1978aa). Notwithstanding this limitation, the NRC staff estimates that constraining the magma partitioning factor to higher values between 0.3 and 0.5 (mean value of 0.4) would imply an increase in calculated doses by a factor of 1.33 compared to results for the full parameter range (between 0.1 and 0.5, with a mean value of 0.3). The NRC staff notes that this small amount of uncertainty on estimated doses due to the magma partitioning factor is offset by the DOE conservative assumption for atmospheric transport that the entire eruption is modeled as violent Strombolian activity in the DOE TSPA. Therefore, the NRC staff notes that the DOE ASHPLUME model reasonably represents the airborne transport of radionuclides and does not underestimate estimated doses for the volcanic ash exposure scenario.

Magma Volume

The NRC staff reviewed the data synthesis and documentation on likely magma volumes for future eruptions provided by DOE in SNL (2007ab) and SNL Section 6.3.4.4 (2007ae). The NRC staff compared the DOE range of eruptive volumes to independent estimates (Jarzemba, 1997aa; Jarzemba, et al., 1997aa; Hill and Connor, 2000aa) and notes that they are in reasonable agreement. The upper end of this range was based on doubling of the Lathrop Wells tephra volume, which was "intended to capture the upper end of the range of uncertainty," and the lower end of the range was based on a calculated tephra fall volume for the smallest cone in the region, Northeast Little Cone, as identified in SNL Section 6.3.4.4 (2007ae) and DOE Enclosure 3 (2009bl). In the ASHPLUME model, DOE used the relationship among eruption power, eruption volume (rather than tephra volume), and eruption duration to constrain the range of total mass of tephra. As discussed in SNL Section 6.5.2.1 (2007ab), DOE constrained eruptive power on the basis of a few observed violent Strombolian eruptions. DOE showed that the tephra-fall volume in the DOE TSPA ranged from 0.004 to 0.14 km^3 [0.001 to 0.03 mi^3] with a mean value of 0.038 km^3 [0.01 mi^3], as identified in DOE Enclosure 3, Figure 1 (2009bl). When considered using the conservative assumption of an entirely violent Strombolian eruption, the NRC staff notes that the resulting range is reasonable because the mass fluxes of magma from the vent are within observed limits.

Ash Particle Diameter

The log-triangular distribution for the mean ash particle diameter {minimum value of 0.001 cm [4×10^{-4} in]}, a mode value of 0.01 cm [4×10^{-3} in], and a maximum value of 0.1 cm [0.04 in] (SNL, 2007ab) and the presented rationale (SNL, 2007ae) are appropriately derived from data obtained at analogous small-volume basaltic volcanoes. The NRC staff considers this particle size range to be representative of available information on violent Strombolian eruptions (e.g., Andronico, et al., 2008aa; Pioli, et al., 2008aa). For comparison, NRC independently assessed basaltic tephra-fall deposits from the 1995 eruption of Cerro Negro (Hill, et al., 1998aa) and determined an average particle diameter of 0.07 cm [0.03 in]. This diameter is within the range of average ash-particle diameters considered by DOE.

Eruptive Power and Duration

DOE analyzed the eruptive parameters of analog volcanoes to develop the range and parameter distribution for eruptive power. The NRC staff compared the DOE eruptive power parameter values to an independent NRC estimate from Leslie, et al. Table 16-1 (2007aa). The DOE range was broader than the independently estimated NRC range, but the majority of the sampled values in the DOE TSPA fall within the NRC range. The geometric means for the two distributions are similar. The DOE parameterization for eruptive power is reasonable because it used analog volcano data to develop parameterization for eruptive power. Duration, when considered with erupted volume, is one indicator of eruptive power, and hence partly controls the height to which the eruption column reaches. The NRC staff notes that DOE's parameterization for eruption duration is reasonable because it is consistent with the range of observations from analog eruptions.

The upper part of DOE's eruptive power range for possible future basaltic eruptions at Yucca Mountain leads to modeled eruption column heights of up to 8.2 km [5.0 mi] (SNL, 2007ab). The NRC staff notes that this upper bound for a violent Strombolian eruption is consistent with results from studies of historic eruptions (Pioli, et al., 2008aa). The NRC staff notes that SAR Sections 2.3.11.1 and 2.3.11.5 mentioned the ability of ASHPLUME to model eruption columns up to 13 km [8.1 mi] high. To build confidence in the ASHPLUME model, DOE exercised this extended upper range for column height to model tephra dispersal for a volcano in New Zealand with a different eruption type, as identified in SNL Appendix J, p. J–23 (2007ab) and compared it with published results on tephra dispersal for that volcano. The NRC staff notes that column heights above 9 km [5.6 mi] are appropriate for modeling the eruption in New Zealand but highlights that such column heights are not appropriate for a violent Strombolian eruption at Yucca Mountain, as identified in SNL Appendix E, p. E–3 (2007ab). DOE clarified that column heights in the ASHPLUME realizations ranged from lower values of about 2 km [1.2 mi] up to a maximum value of 8.2 km [5.0 mi] in the DOE TSPA, as identified in DOE Enclosure 1 (2009bk). Consistent with Jarzemba (1997aa), DOE used observations from analog volcanoes to develop this range of eruptive power. Because the technical basis provided in DOE Enclosure 1 (2009bk) to support this range of column heights shows that the heights did not exceed 8.2 km, the supporting data and treatment of parameter uncertainty are reasonable for modeling a future violent Strombolian-eruption-style event at Yucca Mountain.

Wind Direction and Speed

DOE developed distribution functions for wind speed and wind direction from data provided in National Oceanic and Atmospheric Administration (NOAA) (2004aa). The full range of wind speeds from near zero to the maximum winds observed at the higher altitudes was represented in the wind-speed distribution used in TSPA analyses (SNL, 2007ab). DOE accounted for uncertainty by stochastically sampling wind speed and direction for each eruption realization. DOE Enclosure 1 (2009bk) provided a technical basis by demonstrating that wind speed and wind direction are not correlated at different altitudes in the ASHPLUME model. DOE assigned the same wind speed for the top and lower heights within the column. NOAA (2004aa) showed that for the Yucca Mountain region, wind speeds tend to be greater at higher altitudes (a normal situation); thus, NRC considers that this assignment will not underestimate the dispersal characteristics of a violent Strombolian eruption. This is because NOAA (2004aa) included altitudes up to 13 km [8 mi] and DOE sensitivity analyses in SNL Appendix C (2007ab) and DOE Enclosure 1 (2009bk) model eruption column heights only up to 8.2 km [5 mi]. The NRC staff's review of the DOE sensitivity analyses (SNL, 2007ab) shows that the concentration of tephra at the reasonably maximally exposed individual location after redistribution is relatively insensitive to variable wind conditions during an eruption. The NRC staff notes that the assumption of constant wind speed and direction during an eruption will not affect performance assessment results. For these reasons, the NRC staff notes that DOE provided data and documentation (SNL, 2007ab,ae) to support the selection of parameter ranges for wind speed.

Eruption Column Parameter

Although DOE determined that the column diffusion constant (β) was not an influential parameter, the NRC staff reviewed the DOE use of this parameter in the ASHPLUME model because it controls the vertical distribution of the mass of tephra particles within the eruption column and helps determine the height at which particles exit the column and enter downwind atmospheric transport. The parameter range for β that DOE used is 0.01 to 0.5, and values at the lower end of the distribution lead to more of the tephra mass diffusing (falling) from the eruption column at relatively low altitudes in the modeled eruption (SNL, 2007ab). DOE modeled (DOE, 2009bk) small tephra particle diameters {e.g., 0.005 cm [0.002 in]}, relatively high initial rise velocities {e.g., 9,000 cm/s [3,543 in/s]}, and column diffusion coefficient values (β) less than 0.3 to support upward-concentration particle distributions at realistic heights for violent Strombolian eruption columns. Using the 1995 Cerro Negro eruption in Nicaragua as an analog, the NRC staff performed independent sensitivity analyses for β with a different theoretical model for tephra dispersal. Results showed variations in deposit thickness up to a factor of approximately two over a range of distances on the order of 18 km [11 mi] and beyond (Hill, et al., 1998aa; Winfrey, 2005aa; Janetzke, et al., 2008aa). DOE (2009bk) estimated that varying β from 0.01 to 0.5 reduces the estimated waste concentration by less than 30 percent. On the basis of its review of DOE's result and general agreement with the confirmatory analyses cited previously, the NRC staff notes that DOE's modeling of tephra dispersal from violent Strombolian eruption columns using β values less than 0.3 would not significantly affect dose estimates.

Model Output: Waste Concentration in Tephra

The NRC staff evaluated the outputs from the DOE abstracted model, described in SNL pp. 6–10 (2007ab), on airborne transport for the waste concentration in tephra and its spatial variation with distance and direction from the vent. Although DOE did not determine the waste incorporation ratio to be a significant parameter, NRC evaluated waste incorporation into magma as DOE presented (SNL, 2007ab; SAR Section 2.3.11.4.2) and considered its independent analyses (Codell, 2004aa). The NRC staff notes that the DOE tephra and waste incorporation analysis combined particles according to the compatibility of their respective particle size distributions, consistent with conceptual models of eruptive conduit and fragmentation processes. DOE Enclosure 2 (2009bk) and DOE Enclosures 2 and 3 (2009bm) provided information on the modeled spatial variation of waste concentration in tephra, which demonstrates how much calculated concentrations varied within the same realization (i.e., how the concentration at downwind distances differed from waste concentrations in tephra closer to the vent). On a per-mass basis, waste constitutes a very small fraction of the mass in tephra deposits. Specifically, DOE showed the mass of waste per unit mass of tephra was between 10^{-5} and 10^{-8} in deposits for two representative realizations in DOE Enclosure 2, Figure 1 (2009bk) and DOE Enclosure 3, Supplemental Figure 1 (2009bm). DOE Enclosures 2 and 3 (2009bm) clarified that these values correspond to a single waste package and do not account for the partitioning of waste into scoria cone and lava flows.

In Center for Nuclear Waste Regulatory Analyses Task P–15, p. P15 B–4 (CNWRA, 2007aa), the NRC staff compared these waste concentrations in tephra to independent estimates of waste concentration in tephra from a subset of realizations with the largest values for a single metric ton of waste erupted and accounted for a typical amount of waste entrained in an eruption. NRC staff estimates of waste concentration in tephra deposits are in general agreement with the DOE values. On the basis of the variation in waste concentration provided by DOE, NRC staff notes that variations in waste concentration over the land area representing the location of reasonably maximally exposed individual (i.e., alluvial fan of Fortymile Wash) are not expected to be large within individual realizations. These variations in waste concentration appear to relate to particle size effects on atmospheric transport. This is because, on the basis of the character of tephra deposits in general, smaller sizes of combined tephra and waste particles would represent a greater portion of the deposit at farther distances from the vent, as described in DOE Enclosure 2, Section 1 (2009bk). Therefore, DOE's model reasonably represents the fraction of waste in the tephra deposit.

Model Uncertainty

The NRC staff evaluated model uncertainty in the DOE abstracted model for airborne transport. DOE addressed model uncertainty by considering in SNL (2007ab) several different and widely accepted alternative models, including a Gaussian plume model (PUFF), a gas-thrust code (ASHFALL; Hurst and Turner, 1999aa), and TEPHRA (Winfrey, 2005aa). DOE also evaluated an alternative igneous source term model developed by the NRC staff (Codell, 2003aa) to investigate the processes of waste fragmentation and incorporation into the tephra, and determined that this alternative model was not significantly different from ASHPLUME (SNL, 2007ab), in accord with Codell (2003aa). DOE specifically chose the ASHPLUME model (Jarzemba, et al., 1997aa) because it includes both tephra dispersal and waste incorporation, as used for performance assessment analyses. DOE also accounted for model uncertainty by considering and evaluating several alternative conceptual models for downwind transport of tephra (and waste) from violent Strombolian eruptions. The results of all of these models are in general agreement (SNL, 2007ab).

Model Support

DOE supported its model results with (i) an independent technical evaluation, SNL Appendix E (2007ab); (ii) a comparison to field observations from an analog eruption; and (iii) a comparison to another airborne transport code. With regard to the latter, the ASHFALL code (Hurst and Turner, 1999aa) represents sophisticated models incorporating the physics of tephra transport and deposition but does not include radionuclide transport. ASHFALL uses the same advective-diffusive relationships as ASHPLUME, but employs time- and altitude-dependent wind conditions for tephra dispersal and more explicitly treats tephra particle settling velocities. DOE used the ASHPLUME code in two sets of model runs to reproduce published output from the

ASHFALL code for constant wind conditions and a variable wind field. SNL Appendix J (2007ab) compares the ASHPLUME and ASHFALL model computations, showing that ASHPLUME calculates tephra thicknesses that are within a factor of two of ASHFALL results. DOE also supported its abstracted model with a comparison to field measurements of tephra thickness for the 1995 eruption at Cerro Negro, Nicaragua, outlined in SNL Appendix L (2007ab). The NRC staff reviewed the information cited in items (i) to (iii) of this paragraph and notes that the calculated thicknesses are in reasonable agreement with measured values described in SNL Figure L–6 (2007ab).

Summary of Conclusions for Airborne Transport

On the basis of this evaluation, the DOE ASHPLUME model provides a reasonable estimate for the airborne transport of radionuclides. The documentation provided in SNL (2007ab,ae) described parameterization of the abstracted model.

In TER Section 2.2.1.4.1.3.3.2, the NRC staff evaluates DOE's performance assessment, including the volcanic ash exposure scenario. In that evaluation, four input quantities of the airborne transport abstraction (fraction entrained in ash, tephra volume, tephra density, and ash areal concentration) relate to the representation of the performance assessment for atmospheric transport. These input quantities are considered in the following four paragraphs.

In the DOE model for an extrusive volcanic event, the amount of waste incorporated in tephra scales directly with the magma partitioning factor (SNL, 2007ab). On the basis of the relative proportions of eruptive products at analog volcanoes, DOE selected a range between 0.1 and 0.5 for this parameter, which acts as a direct multiplier on the eruption source term and eruptive dose. The NRC staff notes that a fraction entrained in ash of 0.3 is reasonable for use in the representation calculation in TER Section 2.2.1.4.1.3.2.

DOE analyzed tephra-fall volumes for Quaternary Period (approximately last 2 million years) volcanoes in the Yucca Mountain region by comparison with fall:cone and cone:lava volume ratios for well-preserved young basaltic volcanoes. Violent Strombolian volcanic activity usually yields tephra-fall deposit volumes roughly twice those of the volcanic cone (Hill and Connor, 2000aa). For Lathrop Wells volcano, an appropriate example of the type of eruptive event that could disrupt a repository at Yucca Mountain, the estimated tephra volume is 0.07 km^3 [0.017 mi^3] (SNL, 2007ae). The NRC staff notes that a tephra volume of 0.07 km^3 [0.017 mi^3] is reasonable for use in the representation calculation in TER Section 2.2.1.4.1.3.2.

Bulk *in-situ* density of tephra-fall deposits typically ranges from 0.3 to 1.5 g/cm^3 [0.01 to 0.5 lb/in^3] (Sparks, et al., 1997aa), but is rarely directly measured for basaltic volcanoes. Blong (1984aa) measured a range of tephra deposits that have a density of approximately 1.0 g/cm^3 [0.4 lb/in^3]. DOE (SNL, 2007ae) used 1.0 g/cm^3 [0.4 lb/in^3] for TSPA calculations on the basis of both this value from Blong (1984aa) and a normal distribution of deposit densities ranging from 0.3 to 1.5 g/cm^3 [0.01 to 0.5 lb/in^3] with a mean of 1.0 g/cm^3 [0.4 lb/in^3]. The NRC staff notes that a tephra density of 1 g/cm^3 [0.4 lb/in^3] is reasonable for use in the representation calculation in TER Section 2.2.1.4.1.3.2.

The ash areal concentration was derived from an assumed 1-cm [0.54-in] thickness of deposited tephra. In SNL Appendix G (2007ac), DOE calculated an arithmetic mean of 0.97 cm [0.54 in] for tephra thickness at the reasonably maximally exposed individual location for a wind direction fixed to the south. This result indicates that a 1-cm [0.54-in] thickness is representative of downwind tephra deposits. The NRC staff notes that an ash areal concentration of 10,000 g/m^2 [0.14 lb/in^2] for an assumed 1-cm [0.54-in]-thick deposit is reasonable for use in the representation calculation in TER Section 2.2.1.4.1.3.2.

2.2.1.3.13.3.1.2 Tephra Redistribution in Fortymile Wash

The NRC staff conducted a risk-informed review of tephra redistribution, concentrating on those aspects important to the volcanic ash exposure scenario in the DOE performance assessment, as given next.

Important Aspects of Tephra Redistribution

DOE modeling of redistribution of tephra includes fluvial (running water or stream) transport of contaminated tephra within the Fortymile Wash catchment basin, mixing and dilution with noncontaminated sediment, and deposition of the tephra-sediment mixture on the Fortymile Wash alluvial fan at the location of the reasonably maximally exposed individual. These processes are modeled to occur instantaneously, thus not allowing for any radioactive decay of contaminated tephra before its deposition at the alluvial fan location. In DOE's model, on the alluvial fan, tephra is deposited in distributary channels by redistribution processes and on interchannel divides from airborne transport.

DOE performance assessment results for the volcanic ash exposure scenario are influenced by radionuclide concentrations in soil from both distributary channels and interchannel divides, as described in DOE Enclosure 5, Figure 1 (2009bk). Radionuclides in distributary channels contribute dose to the volcanic ash exposure scenario from the large number of realizations that result in an initial tephra deposit in the Fortymile Wash catchment basin. Fluvial sediment in distributary channels contributes more (two thirds, on average, in the DOE model) to the airborne particle concentration at the reasonably maximally exposed individual location than soils on interchannel divides (one third).

The NRC staff reviewed auxiliary Monte Carlo simulations by Pelletier, et al. (2008aa) that indicated the fluvial transport abstracted model reduced the concentration of tephra in sediment deposited in distributary channels by a factor of about 100 (arithmetic mean of 20 simulations), compared to the tephra concentration in the original tephra deposit. DOE expects any waste attached to tephra particles to remain attached during fluvial transport, and thus expects that any reduction in tephra concentration from fluvial transport should reduce waste concentration by the same amount, as outlined in SNL Section 5.2.5 (2007av). On the basis of its review of DOE's sensitivity analyses described in SNL Section 6.6.1, Figures 6.6.1-1 to 6.6.1-3 (2007av), NRC staff notes that realizations with the largest waste concentrations were most sensitive to critical slope and scour depth, in that order, and slightly sensitive to drainage density. Because DOE included waste dilution during fluvial transport in the FAR model, the NRC staff also focused its review on modeling assumptions and model support.

Summary of Information on Tephra Redistribution

DOE's model of radionuclide redistribution in Fortymile Wash for the volcanic ash exposure scenario was described in SAR Section 2.3.11. In SAR Table 2.3.11-1, DOE identified the features, events, and processes included in the TSPA model.

Following deposition of contaminated tephra (TER Section 2.2.1.3.13.2) from a potential eruption where a volcanic conduit intersects waste packages, DOE's tephra redistribution model accounts for the mobilization of contaminated tephra in the Fortymile Wash catchment basin, dilution of contaminated tephra with uncontaminated sediments in fluvial (stream) channels, and fluvial deposition at the location of the reasonably maximally exposed individual. Fortymile Wash lies east of the repository, which DOE showed to be the most likely direction for tephra dispersal at typical heights for violent Strombolian eruption columns (SAR Figure 2.3.11-15). DOE developed the FAR Version 1.2 code, referred to hereafter as the FAR model, and incorporated this code into its TSPA as a dynamically linked library. The tephra redistribution is abstracted to occur instantaneously (i.e., radionuclide waste transport to the reasonably maximally exposed individual is instantaneous) (SAR Section 2.3.11.4.2.3.1). Eolian (wind-induced) processes are not included in the tephra redistribution model.

In the DOE model described in SNL (2007av), tephra is mobilized and transported downstream if it is initially deposited either on slopes steeper than a critical slope angle or in active channels with stream power exceeding a threshold value. Critical slope parameter values were determined from field measurements at analog sites. DOE determined active channel networks from digital elevation model data and drainage density estimates, on the basis of calibrations to field observations. Channel geomorphology in the Fortymile Wash catchment basin was based on recent observations and is modeled as time invariant. Effects on surface slope, elevation, stream power, and drainage density due to the presence of an initial tephra deposit and its weathering over time were not modeled. DOE considered these effects within the context of existing parametric values and propagated uncertainty, and exclusion of these effects from the model was not expected to significantly change the model results, as described in DOE Enclosure 10, Section 1 (2009bk).

DOE used scour depth estimates to determine the mixing and dilution of tephra with uncontaminated channel sediments. After the 1995 flood event in Fortymile Wash, DOE measured scour depth and estimated a total scour depth to account for the cumulative effect of flood events over time for use in the DOE TSPA. Sediment transport time is not explicitly accounted for in the model for fluvial remobilization and tephra dilution. Instead, a simplification was made such that remobilized tephra would be instantaneously diluted in fluvial sediments and directly deposited at the alluvial fan (i.e., fluvial remobilization, dilution, and deposition occur at the same simulation timestep as initial tephra-fall deposition).

The Fortymile Wash alluvial fan is located at the southern end of the drainage system; DOE modeled it as active (distributary) stream channels and areas between channels (interchannel divides). In the DOE tephra redistribution model, the whole alluvial fan is assumed to be an area occupied by the reasonably maximally exposed individual (SAR Figure 2.3.11-13). Parameter values for the area of the Fortymile Wash alluvial fan and the fraction of that area associated with channels were determined from field measurements and soil geomorphic mapping. In the model of a future volcanic eruption, initial radionuclide concentrations on interchannel divides arise from original tephra-fall deposits across the fan. Redistributed tephra mixed with ambient sediment from the Fortymile Wash drainage system is deposited in distributary channels and not on interchannel divides. Radionuclide concentrations in distributary channels therefore include a mixture of redistributed tephra from the Fortymile Wash drainage system and any original tephra-fall deposits. DOE assumed redistributed tephra is transported as bedload material, which neglects the potential for silt-sized material to be transported in the suspended (streamflow-borne) load past the reasonably maximally exposed individual location and into the Amargosa River Valley. DOE considered alternative modeling approaches during the development and validation of the scour-dilution-mixing approach in its tephra redistribution abstracted model (SNL, 2007av). DOE also referred to model-confidence building, supporting comparisons, and sensitivity analyses documented in SNL (2007av) and a published application of the scour-dilution-mixing model to the area around the Lathrop Wells Volcano (Pelletier, et al., 2008aa).

Time-dependent radionuclide concentrations with soil depth in stream channels and on channel divide surfaces are the ultimate outputs of the tephra redistribution model. TER Section 2.2.1.3.13.3.1.3 evaluates the time-dependent vertical migration of radionuclides in soil for the volcanic ash exposure scenario. In the biosphere model, reviewed in TER Section 2.2.1.3.14.3, the FAR model outputs are combined with biosphere dose conversion factors in the DOE TSPA to estimate annual doses to the reasonably maximally exposed individual (SAR Figure 2.3.10-10).

NRC Staff Evaluation of Tephra Redistribution

The NRC staff reviewed SAR Section 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff's request for additional information (DOE, 2009bk–bm), the supporting DOE information on tephra redistribution presented in SNL (2007av), and information published in peer-reviewed literature (e.g., Pelletier et al., 2008aa).

Model Integration

Model abstractions comprise features, events, and processes that have been screened in from the scenario analysis. In TER Section 2.2.1.2.1.3.1, the NRC staff determines that DOE had identified a list of features, events, and processes for the volcanic exposure scenario, including tephra redistribution. In TER Section 2.2.1.2.1.3.2, the NRC staff determines that DOE had screened all features, events, and processes for the volcanic exposure scenario, including tephra redistribution. DOE did not exclude any features, events, and processes associated with this abstracted model. The NRC staff's review of the tephra redistribution abstracted model evaluates DOE's implementation of the included FEP 1.2.04.07.0C, which accounts for the surface transport processes that redistribute radionuclides following the initial tephra-fall deposition. DOE's technical basis for this abstracted model is given in SNL (2007av).

The following addresses the NRC staff's evaluation of DOE's modeling assumptions used for the fluvial transport in the FAR model.

DOE determined that wetter future climates would increase vegetation on hillslopes and would thus reduce the amount of remobilized tephra from hillslopes into channels, on the basis of the influence of vegetation on erosion. DOE also pointed out that additional precipitation from a wetter climate in the future could increase the scour depth in channels, which was shown in SNL Figure 6.6.1-2 (2007av) to reduce initial radionuclide concentrations in channels. Because neglecting tephra redistribution effects from future wetter climates does not underestimate radionuclide concentrations in sediment, as identified in SNL Section 5.1.1 (2007av), the NRC staff notes that this modeling assumption is reasonable.

The NRC staff notes that DOE's rationale for not explicitly modeling long-term changes to active channels within the depositional fan of Fortymile Wash is reasonable because it is not possible to accurately predict the location and size of future channels in this type of depositional system. For the following reasons, the NRC staff notes that DOE's justification for the area fraction of active channels in the Fortymile Wash alluvial fan and associated parameters in their model is reasonable. The NRC staff evaluated the implementation of the channel area fraction in the FAR model, which also considered its coupling with the biosphere model in the DOE TSPA. DOE assessed (i) the relative susceptibility of the two surfaces, interchannel divides and active fluvial channels, to airborne resuspension and (ii) the assumption that dose contributions from these two surfaces are proportional to their respective fractions of the total area of the alluvial fan. DOE concluded that differences in these two surfaces were accounted for in the DOE TSPA estimates for airborne particle concentration. DOE also showed, in DOE Enclosure 5, Figure 1 (2009bk), that dose contributions for the volcanic ash exposure scenario were much greater from interchannel divides than from channels, for thousands of years after repository closure. Contributions to dose from channels become important after the first 10,000 years as the importance of the inhalation pathway decreased relative to the other biosphere pathways. On the basis of its review of these results, the NRC staff notes that differences in the relative susceptibility of the two surfaces to airborne resuspension, which might have affected their relative dose contributions in addition to their area fraction, would not significantly affect dose estimates. Therefore, the DOE proportional relationship on dose contribution from these two surfaces, on the basis of their respective fractions of the total area of the alluvial fan, is reasonable.

Although DOE acknowledged that a future eruption could alter the channel geomorphology to some degree by deposition of fresh tephra, it assumed that eruption-induced changes would have little effect on the overall geomorphology of the Fortymile Wash catchment basin, as described in SNL Section 5.1.4 (2007av). The NRC staff notes that DOE’s approach, basing fluvial redistribution on current channel geomorphology, is reasonable for modeling future fluvial redistribution and the associated uncertainties in estimating radiological doses. This approach is consistent with information from analog volcanoes such as Parícutin (Segerstrom, 1950aa, 1966aa; Inbar, et al., 1994aa), where original channels were reestablished in the decades following a tephra-fall eruption. SNL Sections 5.1.3 and 6.1.2 (2007av) provide DOE’s justification for not modeling overbank deposition (i.e., deposition of fluvial tephra and sediment on interchannel divides) at the Fortymile Wash alluvial fan. This approach is reasonable because of the absence of fluvial overbank deposits on existing interchannel divides of the Fortymile Wash fan as detailed in SNL Appendix A (2007av).

Rather than accounting for significant rainfall and flooding events individually and tracking the movement of redistributed tephra from each event over time, the FAR model applies a representative deposit for long-term redistribution and dilution of tephra in the same simulation time step. Because FAR model results are integrated with biosphere modeling in the DOE TSPA, the NRC staff considered the coupling of the FAR and biosphere models in the evaluation of FAR model assumptions concerning time dependency. DOE assessed the replenishment of contaminated fluvial deposits over time with respect to time-dependent estimates of resuspended airborne particle concentrations in the DOE TSPA, according to DOE Enclosure 4 (2009bk). DOE clarified that the active outdoor category for reasonably maximally exposed individual activities includes time spent walking outdoors on uncompacted soil or tephra. DOE also acknowledged that airborne particle concentrations would be higher in its TSPA for walking on uncompacted tephra deposits following an eruption than during preeruption conditions. The NRC staff reviewed the DOE parameter ranges for airborne particle concentrations from resuspension following a volcanic eruption and notes that they are supported by published literature referenced in BSC (2006ad). On the basis of its review of information provided [BSC (2006ad); SNL (2007av); DOE Enclosure 4 (2009bk)] and independent field measurements that the NRC staff conducted at analog volcanic sites [Hill, et al. (2000ab); Benke, et al. (2009aa)], the NRC staff notes that DOE’s estimates of inhalation of resuspended particulates from tephra-fall deposits or redistributed tephra in fluvial sediments do not underestimate dose. The NRC staff also notes that the model integration in the DOE TSPA for the volcanic ash exposure scenario, including the coupling of the assumption for instantaneous tephra remobilization in Fortymile Wash with time-dependent values for airborne mass loading, is reasonable to estimate doses over time. Because DOE did not model further mixing and dilution, which could reduce radionuclide concentrations, in the instantaneous deposit from subsequent flooding events in the Fortymile Wash drainage basin, as identified in SNL Section 5.2.4 (2007av), the NRC staff notes that DOE's assumption that drainage system is open (SAR Section 2.3.11.4.4.3) is reasonable. Therefore, modeling a representative deposit for long-term redistribution and dilution of tephra represents the fate of material in this ephemeral drainage system and is a reasonable approach for calculating radiological consequences.

In SNL Section 1.2 (2007av), DOE acknowledged that eolian (wind) sediment transport is a significant geomorphic process in the Yucca Mountain region. DOE clarified in DOE Enclosure 12, Section 1.1 (2009bk) that airborne resuspension of tephra deposits at the reasonably maximally exposed individual location is included in the DOE TSPA as a local-scale eolian redistribution process. The NRC staff notes that the DOE abstracted model did not model long-range eolian redistribution of tephra explicitly. In SNL Section 5.2.2 (2007av), DOE also accounted for potential local-scale eolian transport of contamination in channel sediments onto interchannel divide surfaces by increasing the range for the channel fraction of land area in the Fortymile Wash alluvial fan. As described in SNL Section 5.2.2 (2007av), direct tephra-fall deposition and fluvial processes dominate radionuclide concentrations in the soil and air at the reasonably maximally exposed individual location. DOE considered the long-range eolian transport of freshly deposited tephra south to the location of the reasonably maximally exposed individual from the Fortymile Wash catchment basin to be negligible, on the basis of its characterization of the prevailing direction for strong southerly winds, as identified in SNL Appendix D (2007ab) and Pelletier and Cook (2005aa). DOE also considered relevant wind data in CRWMS M&O Site 9 (1997aa) and determined that southerly winds exhibited higher wind speeds compared to northeasterly winds, described in DOE Enclosure 12, Section 1.2 (2009bk). Higher speeds for south-to-north winds would tend to drive the net transport of contaminated tephra toward the north and away from the reasonably maximally exposed individual location, as DOE Enclosure 12, Section 1.2 (2009bk) identified. DOE therefore concluded that eolian transport of radionuclides deposited in the Fortymile Wash catchment basin to the reasonably maximally exposed individual would be negligible, as stated in SNL Section 5.2.2 (2007av). DOE further concluded that not modeling these eolian effects would tend to overestimate tephra and waste concentration at the reasonably maximally exposed individual location. The NRC staff similarly notes that predicted future south-to-north surface wind directions and associated eolian transport would tend to dilute radionuclide concentrations in the soil and the air at the location of the reasonably maximally exposed individual. The NRC staff thus notes that DOE's technical basis for not explicitly modeling long-range eolian transport of contaminated tephra to the location of the reasonably maximally exposed individual is reasonable. DOE's approach will not underestimate airborne concentrations of radionuclides at that location or dose to the reasonably maximally exposed individual.

Fluvial transport of sediment and tephra in Fortymile Wash is modeled by DOE as bedload transport. From its review of grain-size distributions and textural considerations in SNL Section 5.2.3 (2007av), the NRC staff notes that the DOE modeling approach is reasonable, because bedload transport would be the primary mode of fluvial transport of tephra and sediment in Fortymile Wash. Using grain-size data from analog volcanic eruptions, DOE expects a range of tephra particle sizes with an approximate median value of 0.01–1.0 mm [0.0004–0.04 in], as identified in SNL Section 5.2.3 (2007av). With diameters between 0.002–0.05 mm [0.00008–0.002 in], as described in BSC Section 6.5.3.2 (2006ah), silt-sized particles represent a small portion of this range. DOE considered that silt-sized material could be transported past the reasonably maximally exposed individual location in the suspended load (rather than in the bedload) and concluded that not modeling suspended load transport and deposition is conservative, as stated in SNL Section 5.2.3 (2007av). Because the amount of silt-sized material in the fluvial system is small with respect to the total amount of fluvial material and the DOE FAR model does not exclude tephra or waste associated with silt-sized particles from contributing to dose, the NRC staff notes that DOE's approach for modeling bedload transport and not including other transport mechanisms, such as suspended load, is reasonable.

The NRC staff expects density effects on combined waste-tephra particles (due to the presence of waste) would have a minor to negligible impact on bulk transport, mixing with clean sediment, and waste concentration in redistributed tephra for the large volumes of tephra. This is due to the relatively small ratio of waste concentrations per unit tephra mass, outlined in DOE Enclosure 2, Figure 1 (2009bk) and TER Section 2.2.1.3.10.3.2.3, which supports DOE's conclusion in SNL Section 5.1.5 (2007av) that other processes (physical, chemical, or biological) for concentrating radionuclides at the reasonably maximally exposed individual location are negligible. To enhance confidence in its system description and model integration, DOE had critical reviews performed on an earlier version of the technical basis document and included the review and resolution of comments in SNL Appendix C, Section 7.3.2 (2007av). The NRC staff evaluated the review comments and DOE responses and determined that this process provided additional support for the technical basis used in the FAR model.

Data Sufficiency and Data Uncertainty

The following subsections consider the data sufficiency and propagation of uncertainty presented for the FAR model parameters of critical slope, scour depth, and drainage density, as provided in SNL (2007av).

Critical Slope

DOE collected field data from several analog volcanic sites near Flagstaff, Arizona (i.e., Rattlesnake Crater, Cochrane Hill, Moon Crater, and Cinder Cone), to determine the critical slope parameter range and represent the steepest slope for stable tephra deposits on hillslopes, as outlined in SNL Section 6.5.2 (2007av). DOE assessed the measurement scale of the field observations for critical slope and representativeness of the 30 by 30-m [97 by 97-ft]-grid cell size for the digital elevation map of the Fortymile Wash drainage system. Slopes were measured at a scale of tens of meters [tens to hundreds of feet] in the field, and DOE concluded that the slope angles are representative of hillslopes, in DOE Enclosure 6, Section 1.1 (2009bk). The NRC staff acknowledges that estimating slopes at the scale of an entire hillslope is appropriate for modeling tephra remobilization in the Fortymile Wash drainage system because the potential for tephra mobilization off hillslopes into channels depends on the entire hillslope traversed. DOE clarified that the representation of topography in the FAR model does not smooth steeper slopes and assessed the appropriateness of field data from analog volcanic sites to estimate fluvial erosion of tephra deposits in the Yucca Mountain region. DOE assessed sheetwash and rilling at the field sites (where DOE measured critical slope values) and provided a justification for why rapid postdepositional erosion was not observed and why it is not expected at Yucca Mountain, as described in DOE Enclosure 6, Sections 1.2–1.4 (2009bk). DOE converted field measurement data at analog volcanic sites to its parameter distribution for critical slope, identified in SNL Section 6.5.2 (2007av). The NRC staff notes that DOE parameter uncertainty for critical slope reasonably represents fluvial erosion in Fortymile Wash for postvolcanic conditions.

Scour Depth

SNL Section 6.5.6 (2007av) described the site-specific field data used by DOE to establish the parameter distribution for scour depth: the depth to which water flow will erode (pick up) and move sediment in a stream channel. Parameter values for scour depth were inferred from scour chains, installed by the U.S. Geological Survey (USGS) at the Narrows section of Fortymile Wash, that were subsequently buried by flood sediment about 10 years later in 1995, as shown in SNL Table 6.5.6-1 (2007av). In DOE Enclosure 10, Section 1 (2009bk), DOE explained that other stream flow data (e.g., from the Amargosa Valley station) do not represent tributary conditions of the upper drainage basin. The NRC staff notes that DOE's use of scour measurements that were taken only at the Narrows, for determining scour depth and discharge in Fortymile Wash, is reasonable. Although these scour measurements indicated an upper-bound scour depth of 152 cm [5.0 ft], DOE chose to limit the upper bound of the scour depth parameter to 122 cm [4.0 ft], as identified in SNL Section 6.5.6 and Table 6.5.10-1 (2007av), to add conservatism into the calculation. The average measured scour-depth value from the scour chain data, 73 cm [2.4 ft], was chosen as the lower bound (SNL, 2007av). In SNL Figure 6.6.1-2 (2007av), DOE showed that dilution would be reduced by selecting a shallower scour depth because a smaller amount of uncontaminated sediment would be mixed with contaminated tephra. The NRC staff notes that DOE's interpretation of these field data for scour depth and the selected range for this parameter is reasonable, because restricting scour depth to smaller values overestimates tephra and waste concentrations in fluvial sediment at the location of the reasonably maximally exposed individual.

DOE based its determination of the parameter range for scour depth on site-specific field measurements at Fortymile Wash for current conditions without a surplus of tephra. DOE assessed the potential effect of fresh tephra in Fortymile Wash on scour depth in channels and concluded that scour depth would not be affected by the proportion of tephra in channel sediments. According to DOE Enclosure 10, Section 1 (2009bk), the expected grain sizes of tephra are similar to the observed grain sizes for channel bed material; therefore, different hydraulic conditions were not expected for a fluvial deposit mixing sediment and tephra, as outlined in SNL Section 5.2.3 (2007av) and Pelletier, et al., p. 236 (2008aa). For distances up to a few kilometers [2–4 mi] from a volcanic vent, the NRC staff recognizes that violent Strombolian tephra deposits will consist of particles in the centimeter to millimeter [0.4 to 0.04 in] grain-size range, similar to grain-size distributions in bedload sediment from Fortymile Wash. The NRC staff confirmed that DOE used values for scour depth that tend to underestimate dilution and overestimate dose, described in DOE Enclosure 10, Section 1 (2009bk). The NRC staff notes that the DOE technical basis for scour depth is reasonable and further notes that the scour depth distribution represents posteruption conditions when fresh tephra may be present in Fortymile Wash.

The NRC staff reviewed DOE's scaling approach outlined in SNL Section 6.3.3, Step 3 (2007av) for computing scour depth at different locations in the Fortymile Wash drainage basin. DOE sampled values for the maximum scour depth within the drainage system and computed scour depth at other locations in SNL Eq. 6.3-14 (2007av). SNL Figure 6.3.3-6 (2007av) illustrated the variability of scour depth within the drainage basin. The NRC staff notes that scour depth is dependent on the volumetric water flow (discharge) and acknowledges that contributing area is a common approach for representing water flow at specific locations in the drainage system. Leopold, et al. (1966aa) studied another ephemeral drainage system in a semiarid area of New Mexico and found that most of the erosion occurred in a small percentage of the basin. This observation is consistent with the variability in scour depth presented in SNL Figure 6.3.3-6 (2007av). DOE's scaling approach for scour depth is reasonable because the variability in scour depth is based on estimates of slope angle [described in SNL Figure 6.3.3-3 (2007av)], contributing area [outlined in SNL Figure 6.3.3-5 (2007av)], and stream order (Leopold, et al., 1964aa) in the drainage basin. The NRC staff notes that the greatest scour depth is expected to occur where the main drainage channel narrows, because scour depth is dependent on volumetric water flow and channel width.

Drainage Density

The drainage density is the ratio of the total length of streams to the area of the drainage system (length per unit area). Compared to critical slope and scour depth, performance assessment outputs are less sensitive to drainage density for the volcanic ash exposure scenario. DOE estimated drainage density from simulations of 34 channel heads on the eastern slope of Yucca Mountain. In SNL Eq. 6.3-7 and p. 6-16 (2007av), DOE used the reciprocal of the drainage density as a stream power threshold for determining active channels within the Fortymile Wash catchment basin; in SNL Section 6.5.6 and Figure 6.5.6-3 (2007av), DOE compared modeled channel head locations to actual locations and selected the drainage density that yielded the smallest average distance difference. DOE assigned the parameter range for drainage density by considering other values that provided agreement between observed and calculated channel heads on Yucca Mountain. This approach is reasonable for determining the drainage density because it is based on a site-specific comparison and an established scientific relationship for stream power and contributing areas. The NRC staff notes that the resulting parameter range for drainage density adequately accounts for uncertainty because it is based on a site-specific comparison and a reasonable interpretation of the results of the comparison.

DOE assigned values to the previously mentioned parameters (critical slope, scour depth, and drainage density) in the Fortymile Wash Ash Redistribution (FAR) model according to the 30 by 30-m [97 by 97-ft] grid cells on the digital elevation (topographic) map of the Fortymile Wash drainage system. In the DOE TSPA analysis, sampling of these parameters is performed in the FAR model independently, without correlation between them [SNL Table 6.5.10-1 (2007av)]. The NRC staff notes that representing topography and associated surface processes in the FAR model at a 30 by 30-m [100 by 100-ft] scale is a reasonable approach, because a strong correlation between critical slope, scour depth, and drainage density is not expected, as these parameters characterize different aspects of the system (i.e., material stability on hillslopes, intrachannel flow and mixing, and active channel threshold within the network, respectively). With this approach, the model will not underpredict the consequences and any alternative approach will not significantly affect the results.

Model Uncertainty

The NRC staff reviewed the DOE consideration of alternative models in Pelletier, et al. (2008aa) and SNL Sections 6.2.2, 6.3.3, and 7.2.4 (2007av). As previously described in TER Section 2.2.1.3.13.3, the NRC staff determined that the DOE abstracted model for fluvial transport and tephra dilution significantly reduces radionuclide concentration and influences the results of the volcanic ash exposure scenario. The FAR model is based on a new modeling approach for the natural processes of scour, dilution, and mixing developed by DOE specifically for the Fortymile Wash drainage system. The classic dilution-mixing model has been generally considered the standard approach (Hawkes, 1976aa; Marcus, 1987aa) in the past. In DOE Enclosure 7, Section 1 (2009bk), DOE considered the classic dilution-mixing model as appropriate only for tributary systems that discharge into the sea or a lake but not well suited for Fortymile Wash, because it is a tributary-distributary inland drainage system in a desert region. DOE also identified other shortcomings with classic dilution-mixing models, such as the inability to model the vertical distribution of contamination and dilution of contaminated with uncontaminated sediments. Because DOE considered scour-dilution mixing as the predominant mode of dilution, it concluded in DOE Enclosure 7, Section 1.1 (2009bk) that the scour-dilution-mixing model more accurately represented the processes at Fortymile Wash. DOE further concluded in DOE Enclosure 7, Section 1.2 (2009bk) that dilution-mixing models were not directly applicable.

The NRC staff notes that the FAR model reasonably represents the Fortymile Wash drainage system and that DOE considered alternative conceptual models to the scour-dilution-mixing approach used in the FAR model. The NRC staff notes that the FAR model would not underpredict the radiological consequences to the reasonably maximally exposed individual and that it is supported by applicable data.

Model Support

The NRC staff evaluated the model support for the FAR model. DOE supported its model results with (i) independent technical evaluations, as outlined in SNL Appendix C (2007av), and (ii) a peer-reviewed journal article (Pelletier, et al., 2008aa) that included a site-specific comparison for fluvial redistribution and dilution of tephra from the Lathrop Wells volcano. The NRC staff reviewed the independent technical evaluators' comments and notes that DOE responded to and resolved comments. The NRC staff also reviewed Pelletier, et al. (2008aa) and agrees with the article's conclusion that a scour-dilution-mixing approach is suitable for estimating downstream contamination concentrations when bedload transport dominates and overbank sedimentation is not significant. DOE used a modeling assumption to constrain the tephra thickness in a channel routed to the Fortymile Wash alluvial fan to not exceed the scour depth, as described in SNL p. 6-24 (2007av). In SNL Section 7.3.1.2 (2007av), DOE also pointed to the 77,000-year-old analog volcano at Lathrop Wells for observations of long-term storage of tephra below the scour depth. Although tephra stored below the scour depth in connected channels (i.e., connected to the main Fortymile Wash channel) or in unconnected channels cannot contribute to the concentration of radionuclides at the Fortymile Wash alluvial fan, total tephra stored below the scour depth was estimated by DOE to be on the order of 3 to 7 percent of the amount of tephra-fall in the Fortymile Wash drainage system, as identified in DOE Enclosure 11 (2009bk). Because increases in scour depth tend to result in greater dilution of tephra with uncontaminated sediment in the DOE FAR model, the NRC staff notes that tephra storage below the scour depth does not significantly affect the estimates of radionuclide concentrations in sediment at the location of the reasonably maximally exposed individual. The NRC staff acknowledges that sediment storage in channels is a well-recognized geomorphological phenomenon and notes that the DOE model for fluvial transport and dilution of tephra in Fortymile Wash has been supported.

NRC Staff Evaluation of Tephra Redistribution

On the basis of the evaluation described previously, the DOE FAR model provides a reasonable approach for calculating the redistribution of tephra to the reasonably maximally exposed individual location. DOE integrated its model of tephra redistribution by incorporating the important processes associated with the ash redistribution via soil and sediment transport feature, event and process. The NRC staff notes that the parameter determinations for the redistribution of tephra and incorporated waste in DOE's abstracted model for fluvial transport in the Fortymile Wash catchment basin are described and justified. The NRC staff notes that the data support this abstracted model in the TSPA. The NRC staff notes that data uncertainty was characterized and propagated through the abstracted model for fluvial transport in Fortymile Wash. Parameter ranges were described and justified. The NRC staff notes that DOE's treatment of alternative conceptual models is reasonable because no alternatives that are consistent with the presented data would affect the results significantly. Because DOE supported its model results with independent technical evaluations and a site-specific comparison for fluvial redistribution and dilution of tephra from the Lathrop Wells volcano, the NRC staff notes that DOE's model is supported.

2.2.1.3.13.3.1.3 Downward Migration of Radionuclides in Soil

The NRC staff conducted a risk-informed review of the model of downward migration of radionuclides in soil, concentrating on those aspects important to the volcanic ash exposure scenario in the DOE performance assessment, as given next.

Important Aspects of Downward Migration of Radionuclides in Soil

In the DOE TSPA, both long- and short-term inhalation significantly contribute to the total dose for the volcanic ash exposure scenario for 100,000 years. Because the short-term inhalation contribution is dominated by a much faster rate of reduction in airborne mass loading, the vertical migration of radionuclides has greater potential influence on long-term inhalation dose. As previously discussed, contributions to total dose from the inhalation of particulates diminish after 100,000 years.

The DOE results indicated that processes contained in this abstracted model for the volcanic ash exposure scenario result in a small reduction in radionuclide concentrations over time. The NRC staff obtained quantitative insights by investigating intermediate output files from the DOE TSPA. For unplowed soil, the reduction of radionuclide concentration due to vertical migration out of the resuspendable layer is gradual and slows with increasing time following initial deposition. On average, the radionuclide concentration in the resuspendable layer required approximately 20, 150, 700, and 4,000 years to decrease by a factor of 2, 4, 8, and 16, respectively, from its initial value in the DOE TSPA. For plowed soil, radionuclides are uniformly mixed within the tillage depth, and the time-dependent reduction in radionuclide concentration due to vertical migration is small (reduction by a factor of about 2 in 10,000 years). For fluvial channels, radionuclides are assumed to be well mixed within the fluvial sediment deposit, and the time-dependent reduction in concentration due to vertical migration of radionuclides out of either the resuspendable layer or tillage depth is minimal (leading to a reduction of less than a factor of 2 in 10,000 years). Sensitivity analyses performed by DOE indicated that radionuclide concentration is most sensitive to the diffusivity rate in soil on interchannel divides, followed by a lesser sensitivity to the diffusivity rate in fluvial channels. There was a negligible sensitivity to different values of permeable depth on the interchannel divides, as described in SNL Section 6.6 (2007av).

Summary of Information on Downward Migration of Radionuclides in Soil

Modeling of radionuclide concentration migration into soil for the volcanic ash exposure scenario was described in SAR Sections 2.3.10, Biosphere Transport and Exposure, and 2.3.11, Igneous Activity. In SAR Tables 2.3.10-1 and 2.3.11-1, DOE identified the features, events, and processes included in the TSPA model.

Calculation of radionuclide concentrations with soil depth at the reasonably maximally exposed individual location is one of the main elements of the DOE FAR model for tephra redistribution.

DOE developed this part of the FAR model specifically for the Fortymile Wash alluvial fan, consisting of active channels and interchannel divide surfaces. The exposure of the reasonably maximally exposed individual to radionuclides in soil was modeled for two layers: (i) a thin upper surface layer from which particles can be suspended into the atmosphere by disturbances and (ii) a thicker, lower surface layer that may undergo mixing by agricultural practices such as tillage (SAR Section 2.3.10.2.6).

The FAR model includes the downward migration of radionuclides into soil for the volcanic ash exposure scenario. Incorporated into the DOE TSPA as a dynamically linked library, the FAR model is connected to the surface soil submodel of the DOE biosphere model, Environmental Radiation Model for Yucca Mountain Nevada (ERMYN), which calculates biosphere dose conversion factors for the volcanic ash exposure scenario on the basis of unit concentrations of radionuclides in volcanic ash deposited on the ground. The surface soil submodel is included in the biosphere analysis of all exposure pathways for the volcanic ash exposure scenario (refer to SAR Figures 2.3.10-8 and 2.3.10-10). Biosphere dose conversion factors are combined with time-dependent radionuclide concentrations in soil from the FAR model to estimate annual doses for the volcanic ash exposure scenario.

DOE modeled time-dependent vertical migration of radionuclides in soil within the FAR tephra redistribution model as a diffusive process in one dimension. Values for radionuclide diffusivity and permeable depth differed between those areas on interchannel divides and those in fluvial channels. Field data on Cs-137 concentration profiles from the upper Fortymile Wash alluvial fan were used to determine radionuclide diffusivities and the associated uncertainties.

Permeable depths in soils were determined from field measurements in pits dug on interchannel divides of the Fortymile Wash alluvial fan and from USGS data on scour depth in fluvial channels. Although advection is not explicitly modeled, DOE identified that diffusivity data accounted for all transport mechanisms, including advection and bioturbation. DOE does not include effects of future climate change on the modeled processes and parameters in the tephra redistribution model, because DOE concluded that processes associated with future climate change would only decrease radionuclide concentrations in soils (SAR Section 2.3.11.4.4.3). In the FAR model, radionuclides are restricted from migrating into a deeper horizon by use of a reduced permeability. The reduced permeability was assumed to be caused by a greater carbonate or clay content than the minor content in surface and near-surface soils. DOE's approach limited possible reduction of radionuclide concentrations in the surface layer due to vertical migration over long time periods.

For the volcanic ash exposure scenario, the surface soil submodel calculates radionuclide mass concentrations in the tilled surface soil layer and in the thin resuspendable layer for noncultivated soil. In the DOE TSPA, radionuclide concentrations in the resuspendable layer and tilled soil are applied to different environmental exposure pathways. Weighting factors for land usage (e.g., fractions of land that are tilled and not tilled) are not included in the dose calculations. Igneous eruption dose calculations include weighting factors for the fraction of land area apportioned into active fluvial channels and interchannel divides. Volcanic material (basaltic tephra) is assumed to be mixed uniformly in tilled surface soil. Concentrations of radionuclides in tilled surface soil are factored into the pathway analysis for ingestion of contaminated crops and animal products. Inhalation and external exposure pathway calculations are dependent on radionuclide concentrations in the resuspendable layer. Because erosion and other surficial processes are accounted for in the tephra redistribution model, DOE's surface soil submodel does not include these processes for the volcanic ash exposure scenario.

NRC Staff Evaluation of Downward Migration of Radionuclides in Soil

The NRC staff reviewed SAR Sections 2.3.10 and 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff's request for additional information (DOE, 2009bk), and the supporting DOE information on tephra redistribution presented in SNL (2007av).

Model Integration

Model abstractions comprise features, events, and processes that have been screened in from the scenario analysis. In TER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE identified a complete list of features, events, and processes for the volcanic exposure scenario, including downward migration in soil. In TER Section 2.2.1.2.1.3.2, the NRC staff notes that DOE had screened all features, events, and processes for the volcanic exposure scenario, including downward migration in soil. DOE did not exclude any features, events, and processes associated with this abstracted model. The NRC staff's review of the abstracted model for downward migration in soil evaluates DOE's implementation of the included features, events, and processes: (i) FEP 1.2.04.07.0C, (ii) FEP 2.3.02.01.0A, (iii) FEP 2.3.02.02.0A, and (iv) FEP 2.3.02.03.0A.

As discussed further in the next sections, the NRC staff notes that the DOE TSPA analysis incorporates important features, events, and processes and couplings between different models associated with the vertical migration of radionuclides in soil for the volcanic ash exposure scenario. TER Section 2.2.1.3.13.1.2 evaluates the treatment of climate in the DOE FAR model. The technical basis for this abstracted model was described in SNL (2007av); modeling assumptions were described in SNL Section 5 (2007av). DOE neglected effects due to future wetter climates in this abstracted model on the basis that wetter climates could increase radionuclide diffusivities in soil, increase vertical migration, reduce the radionuclide concentrations in surface soil layers, and thus reduce estimated doses. This DOE modeling approach is reasonable because wetter climates would likely increase radionuclide migration to deeper soil layers, as identified in Till and Moore Eq. 2 (1988aa), and thus result in lower estimated doses. Therefore, the NRC staff notes that neglecting the potential effects of a future wetter climate on radionuclide migration would not underestimate dose.

The NRC staff evaluated critical modeling assumptions for this abstracted model. DOE assumed that all radionuclides migrate into the soil at the same rate, because in temperate climates, weathering of radionuclides from the soil surfaces into deeper soil layers is mainly a physical, rather than a chemical, process. This approach is supported by the similar rate of radionuclide migration for radionuclides of different chemical characteristics (Anspaugh, et al., 2002aa). NRC considered the small reduction of radionuclide concentrations over time credited in the DOE model and notes that the field data support the reduction presented in SNL (2007av).

The DOE assumption for not explicitly modeling advection (i.e., flow by liquid movement) of radionuclides is reasonable because including advective transport would likely increase the removal of radionuclides from the soil surface and thus reduce calculated doses. In addition, independent critical reviews from non-DOE, academic-based experts were conducted on an earlier version of the technical basis document, and DOE included the review comments and responses in SNL Appendix C, Section 7.3.2 (2007av). The NRC staff reviewed those independent technical evaluations and notes that DOE responded to and resolved comments by independent reviewers that pertained to the vertical migration of radionuclides in soil.

Data Sufficiency

The NRC staff evaluated data sufficiency in the DOE abstracted model for the downward migration of radionuclides in soil. This abstracted model consists of parameters for permeable depth in fluvial channels and on interchannel divides, soil diffusivity of radionuclides in channels and on interchannel divides, and land fraction of the Fortymile Wash alluvial fan attributed to channels. Parameter values for permeable depth in channels were inferred from USGS data on scour depth in channels and the previously mentioned site-specific field data from soil pits, as described in SNL Section 6.5.5.2 (2007av). Diffusivity rates for radionuclide migration in soils on interchannel divides and in fluvial channels were determined from site-specific field data of Cs-137 profiles. These profiles resulted from contaminated fallout deposited approximately 50 years earlier from atmospheric nuclear weapons testing. DOE performed soil-geomorphic mapping of the Fortymile Wash alluvial fan to determine the fraction of the fan area that has been subjected to fluvial erosion and deposition within the past 10,000 years, as identified in SNL Appendix A (2007av). Diffusivity rates for fluvial channels were determined from measurements from surfaces DOE characterized as active channels. Measured data from older terraces were used to calculate the diffusivity rate for interchannel divides. On the basis of these two field data sets, DOE specified separate diffusivity parameters for older surfaces of the interchannel divides and younger surfaces in fluvial channels. Because the time-dependent reduction of radionuclide concentrations in soil is slow in the DOE TSPA, the NRC staff notes that modeling of the downward migration of radionuclides tends to overestimate dose to the reasonably maximally exposed individual. The NRC staff notes that the data support this abstracted model in the TSPA because they are from site-specific field observations. These parameter determinations are described and justified. For these reasons, the NRC staff notes that the DOE use of measured diffusivity rates for cesium in soil is reasonable for modeling the reduction of radionuclide concentrations with time in the DOE TSPA for the volcanic ash exposure scenario.

Data Uncertainty

In the DOE abstracted model for the vertical migration of radionuclides in soil, parameter distributions are applied to account for data uncertainty. As discussed in the preceding paragraphs, the NRC staff reviewed these parameter distributions and notes that the range of uncertainty in these parameters is consistent with the technical basis used to develop the parameters. A permeable depth in fluvial channels of 200 cm [79 in] was derived from field measurements and is used as a constant value, as outlined in SNL Section 7.1.3 and Table 4.1-4 (2007av). DOE supported this constant value with an argument that the permeable depth in fluvial channels could be much deeper. The NRC staff notes that neglecting deeper permeable layers is conservative because increases in permeable depth reduce long-term radionuclide concentrations in soil. Given the minimal effect that vertical migration has on radionuclide concentrations in fluvial channels over time, this treatment of uncertainty in permeable depth is reasonable.

Model Uncertainty

The NRC staff evaluated model uncertainty for the downward migration of radionuclides in soil following an eruption. For conditions after a potential future volcanic eruption that intersects the repository and entrains waste, radionuclides on the ground surface would originate as radionuclide contamination in basaltic tephra deposits, as discussed previously. DOE used site-specific data from the deposition and migration of fine radionuclide particulates into current surface soils of the Fortymile Wash alluvial fan, which are not rich in basaltic material, to support its model for the downward migration of radionuclides following deposition in the volcanic ash exposure scenario (SAR Section 2.3.11.4.2.3; SNL, 2007av). DOE provided a technical basis for neglecting the effects of fresh basaltic tephra on radionuclide diffusion in soil. The technical basis for radionuclide migration was provided in two parts: one for channel sediments and the other for soils on interchannel divides. For channel sediments, DOE used field observations at Lathrop Wells, described in SNL Section 7.3.1.1 (2007av), to show that basaltic and nonbasaltic sediments in the drainages exhibited similar grain sizes and transport rates. DOE also found basaltic and nonbasaltic sediments to be well mixed. DOE reported a significant amount of dilution of fresh basaltic tephra with nonbasaltic sediments during fluvial transport in the Fortymile Wash drainage basin. In particular, tephra concentrations in channel sediments were less than 20 percent at the reasonably maximally exposed individual location in the DOE TSPA. For these reasons, DOE concluded that determining separate diffusion rates of radionuclides in basaltic tephra was not necessary for estimating the downward migration of radionuclides in mixed channel sediments.

The NRC staff notes that the DOE diffusion model represents uncertainty for radionuclide migration in channel sediments for the volcanic ash exposure scenario. For soils on interchannel divides, DOE concluded in DOE Enclosure 3 (2009bk) that differences in diffusivity for a basaltic tephra deposit on ambient soils would be negligible because tephra thicknesses at the reasonably maximally exposed individual location would be thin {less than 0.33 cm [0.85 in] for about 90 percent of TSPA simulations with a primary tephra-fall deposit near the reasonably maximally exposed individual location} and grain-size ranges for tephra and ambient soils on interchannel divides are similar. The NRC staff notes that the DOE diffusion model does not permit radionuclides to migrate below the lower boundary of the surface soils, defined by parameters for the permeable depth. For very long times after the volcanic event, radionuclide concentration profiles in surface soil layers become uniformly distributed with depth in the DOE model. As previously discussed in TER Section 2.2.1.3.13.3.1.3, the NRC staff notes that this assumption is reasonable because neglecting deeper permeable depths would reduce the long-term radionuclide concentrations in soil. DOE's parameter ranges and selected distributions are reasonable because they represent expected conditions of the Yucca Mountain region for the volcanic ash exposure scenario.

DOE did not identify any alternative conceptual models that would likely affect the timing or magnitude of dose. The DOE model slowly distributes radionuclides from surface layers to deeper layers and restricts radionuclides from migrating below permeable soil layers. The NRC staff notes that the DOE diffusion model will not underestimate radiological dose. The NRC staff also notes that the approach DOE used is conservative because there are no alternative models consistent with available information that would significantly increase radiological dose.

Model Support

The NRC staff evaluated model support for the downward migration of radionuclides in soil. DOE used a one-dimensional diffusion model for the downward migration of radionuclides in soil with measurements of Cs-137 radioactivity profiles in soils at Fortymile Wash. Pelletier, et al. (2005aa) published a peer-reviewed journal article that supported use of a diffusion model for radionuclide migration in soil at the Fortymile Wash alluvial fan. The article also included the synthesis of field data for determining radionuclide diffusivities in soil at the fan. In SNL Section 7.2.6 (2007av), DOE verified the appropriateness of its diffusion model for characterizing the radionuclide concentration profiles in soil. DOE supported its abstracted model with a geomorphic characterization of the Fortymile Wash alluvial fan, as identified in SNL Appendix A (2007av), and derivation of its mathematical model for diffusion, described in SNL Appendix E (2007av). For the aforementioned reasons, DOE has provided reasonable support for its diffusion model for radionuclide migration in soil following an igneous eruption.

NRC Staff Evaluation of Downward Migration of Radionuclides in Soil

On the basis of the evaluation described previously, the NRC staff notes that the DOE FAR model provides a reasonable approach for calculating the downward radionuclide migration in the soil at the location of the reasonably maximally exposed individual. DOE integrated its model of downward radionuclide migration in soil by incorporating the important processes associated with the four included features, events, and processes. The parameter determinations and their uncertainties for the downward radionuclide migration are reasonably described and justified. The NRC staff notes that the data support this abstracted model in the TSPA because they are from site-specific field observations. The NRC staff notes that data uncertainty was characterized and propagated through the abstracted model for downward radionuclide migration in soil. DOE's treatment of alternative conceptual models is reasonable because no alternatives that are consistent with the presented data would affect the results significantly. The NRC staff also notes that DOE supported its abstracted model with a geomorphic characterization of the Fortymile Wash alluvial fan and derivation of its mathematical model for diffusion.

Summary of NRC Staff Evaluation of the Volcanic Ash Exposure Scenario

The NRC staff has reviewed the information in the SAR and supporting information. DOE provided information for the airborne transport of radionuclides, tephra redistribution in Fortymile Wash, and downward migration of radionuclides abstracted models, and also for their implementation in the GoldSim modeling environment of the TSPA analysis. The SAR considered data from the site and surrounding region, considered uncertainties and variability in parameter values, and used alternative conceptual models in the analyses. Specific features, events, and processes have been included in the SAR, and technical bases have been provided for inclusion or exclusion of features, events, and processes. The SAR included specific degradation, deterioration, and alteration processes, and the effects of these processes were considered in evaluating annual dose. The SAR included technical bases for models used in the performance assessment for time periods after 10,000 years and through the period of geologic stability.

2.2.1.3.13.3.2 Assessment and Review of Groundwater Exposure Scenarios

For the groundwater exposure scenario, the surface soil submodel addresses the vertical movement of radionuclides in the soil that follows from irrigation with contaminated groundwater (SAR Figure 2.3.10-1) and calculates a time-dependent profile of radionuclide concentration in the contaminated soil horizon at the reasonably maximally exposed individual location. Radionuclide contamination in groundwater can result from waste package failure due to corrosion, mechanical disruption, or potential disruption by intruding magma. Radionuclide contamination in groundwater serves as input to the surface soil submodel. TER Section 2.2.1.3.12 documents the NRC staff's review of the DOE approach to estimating radionuclide contamination in groundwater. This section addresses the vertical movement of radionuclides in the soil from contaminated groundwater irrigation together with background precipitation. As described next, DOE's results indicated the influence of the surface soil submodel on the DOE calculated repository performance.

The NRC staff conducted a risk-informed review of DOE's surface soil submodel using YMRP Section 2.2.1.3.13. The NRC staff reviewed the important aspects of the groundwater exposure scenario in the DOE performance assessment.

Important Aspects of the DOE Surface Soil Submodel

The NRC staff reviewed the SAR and assessed the extent to which the DOE surface soil submodel influences the DOE calculation of repository performance. The surface soil submodel is used to estimate radionuclide doses for the groundwater exposure scenarios, including the seismic ground motion and igneous intrusion scenarios. SAR Figure 2.4-18 (a and b) showed that the seismic ground motion and igneous intrusion scenarios dominate the estimated total mean annual dose for 10,000 and 1 million years after repository closure. Total doses from the groundwater exposure scenarios are controlled by multiple radionuclides and dose pathways. Because the surface soil submodel is a component of the DOE biosphere model ERMYN (SAR Figure 2.3.10-9; BSC, 2006ah; SNL, 2007ac), its importance within the DOE TSPA depends on specific exposure pathways and radionuclides.

SAR Figures 2.4-20 and 2.4-30 indicated that a combined set of radionuclides—C-14, Tc-99, I-129, Ra-226, Np-237, Pu-238, Pu-240, and Pu-242—can contribute significantly to total dose over time. The NRC staff used this DOE information and radionuclide set to assess the influence of the surface soil submodel on TSPA results. Pathways linked to the surface soil submodel (radon inhalation and external exposure) account for more than 80 percent of the Ra-226 biosphere dose conversion factor, as described in SNL Tables 6.13-1 and 6.13-2 (2007ac). Pathways linked to the surface soil submodel account for approximately 50 percent of the Tc-99, Pu-239, Pu-240, and Pu-242 biosphere dose conversion factors, as identified in SNL Tables 6.13-1 and 6.13-2 (2007ac). In SNL Tables 6.13-1 and 6.13-2 (2007ac), pathways linked to the surface soil submodel accounted for less than 35 percent of the Np-237, I-129, and C-14 biosphere dose conversion factors. On an individual radionuclide basis, the DOE surface soil submodel can have a large influence on estimated doses from Ra-226, a moderate influence on doses from Tc-99 and various plutonium isotopes, and a small influence on other radionuclide doses. Because no single, dominant radionuclide exists and Ra-226 contributes only a fraction to the total dose [SAR Figure 2.4-20(b)], NRC staff notes that these DOE results indicate the limited influence of the DOE surface submodel on the DOE-calculated repository performance.

Summary of Information on the Surface Soil Submodel

The surface soil submodel is one component of the DOE biosphere model, which is described in SAR Section 2.3.10, Biosphere Transport and Exposure. In SAR Table 2.3.10-1, DOE identified the features, events, and processes included in the TPSA model.

The surface soil submodel calculates the radionuclide concentrations in both cultivated field and garden surface soils following radionuclide release in the groundwater pathway. The output from the surface soil submodel serves as input for various biosphere submodels (animal, ingestion, external, plant, and air). The outputs of the biosphere model are biosphere dose conversion factors, which are factors that provide the dose per unit concentration in a medium such as water, for groundwater exposure (SAR Figure 2.3.10-9). Biosphere dose conversion factors are combined with the time-dependent radionuclide concentrations in groundwater from the saturated zone transport models to calculate annual dose to the reasonably maximally exposed individual from groundwater exposure (SAR Figure 2.3.10-9). Results from the surface soil model are used to determine potential doses from inhalation of suspended soil particles, consumption of radionuclide-containing crops, soil ingestion by humans and animals, exposure to radioactive gases from the surface soil, and external exposure to radionuclide-containing soils. TER Section 2.2.1.3.14 documents the NRC staff's evaluation of the biosphere dose conversion factors and biosphere submodels other than the surface soil submodel.

In the surface soil submodel, radionuclides are considered to be added to the soil from irrigation using contaminated groundwater. They may decrease through the mechanisms of radioactive decay, leaching into deeper zones, erosion of soil particles, and gaseous release to the atmosphere (i.e., radon and carbon dioxide). Two soil layers are considered: a thin upper surface layer from which particles can be suspended into the atmosphere by disturbances and a thicker lower surface layer that may undergo mixing by agricultural practices such as tilling the land.

NRC Staff Evaluation of the Surface Soil Submodel

The NRC staff reviewed SAR Sections 2.3.10 on the surface soil submodel and the supporting DOE information on the surface soil submodel presented in SNL (2007ac) and BSC (2006ah).

Model Integration

Model abstractions comprise features, events, and processes that have been screened in from the scenario analysis. In TER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE had identified a complete list of features, events, and processes for the groundwater exposure scenario. DOE screened out the transport of radionuclides past these soil layers to greater depths in its analysis of features, events, and processes. In TER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE had screened all features, events, and processes for the groundwater exposure scenario, including those associated with the surface soil submodel. The NRC staff's review of the downward migration modeling in soil evaluates DOE's implementation of the included features, events, and processes: (i) FEP 2.3.02.01.0A, (ii) FEP 2.3.02.02.0A, and (iii) FEP 2.3.02.03.0A.

DOE considered two soil layers in the surface soil submodel: (i) a thin surface layer that is susceptible to particles being suspended in the atmosphere from disturbances such as agricultural activities and wind and (ii) a lower, thicker layer that is approximately the thickness of the plow or till zone. Radionuclide concentrations for primary radionuclides and two long-lived decay products are calculated for varying climate conditions. Radionuclide concentrations are assumed to be uniform in the thin resuspendable layer and uniform in the thicker surface layer, if tilling is practiced.

The NRC staff evaluated the modeling assumptions and integration for the surface soil submodel. Radionuclides in the surface soil submodel originate from contaminated groundwater used for irrigation. DOE used a unit concentration for each radionuclide of 1 Bq/m^3 in the irrigation water to determine normalized biosphere dose conversion factors. TSPA computes the radionuclide doses by multiplying these normalized factors by the radionuclide concentrations in the groundwater. DOE allowed the radionuclide concentration absorbed on soils to build up toward equilibrium conditions before estimating the potential dose to the reasonably maximally exposed individual. NRC staff notes that once equilibrium conditions are obtained, longer irrigation periods would not affect radionuclide concentrations in soil. So that the potential dose to the reasonably maximally exposed individual at earlier times would not be underestimated, DOE determined radionuclide concentrations in soil by assuming irrigation with contaminated well water for periods up to 1,000 years before estimating the potential dose to the reasonably maximally exposed individual. This approach to the prior irrigation period and buildup of radionuclides is reasonable because the use of higher radionuclide concentrations in the surface soil would tend to overestimate dose.

The mathematical model DOE used, outlined in SNL Eq. 6.4.1-1, p. 6-73 (2007ac), to represent addition and removal of radionuclides in the surface soil is a differential equation that considers the dominant governing processes. The differential equation relates the rate of radionuclide accumulation to the difference between the rate of radionuclide addition and the rate of radionuclide loss in a volume of soil. This type of differential equation is known as an equation of continuity and is commonly used to track mass movement through systems (Bird, et al., 1960aa). Inherent in the equation is mass balance that accounts for the difference between radionuclide addition and removal per unit time. The differential equation accounts for changes in storage or radionuclide concentration in the soil's surface. Although the mathematical model and associated parameters DOE used do not account for all phenomena at a small (pore-level) scale, the NRC staff notes that the overall behavior at larger scales, for which the parameters are justified, is represented because small-scale phenomena are captured in the parameter determination. This mathematical model describes radionuclide movement at a scale where parameter values do not vary significantly with relatively small changes in spatial scales. For analyses evaluating potential doses to the receptor (reasonably maximally exposed individual), the NRC staff notes that this is a reasonable modeling approach. For the reasons described previously, the conceptual and mathematical surface soil submodels are reasonable for determining average radionuclide concentrations in the surface soil resulting from irrigation with radionuclide-containing groundwater.

Radioactive decay, transport (i.e., leaching) to deeper soil, erosion of soil particles, and gaseous release to the atmosphere of radon and carbon dioxide are the dominant mechanisms for removal of radionuclides from the surface soil layers. Short-lived radioactive decay products (i.e., having half-lives shorter than 180 days) are assumed by DOE to be in equilibrium with the long-lived primary radionuclides. The NRC staff acknowledges this assumption as a common approach in environmental modeling and notes that it will not underestimate the effects of short-lived radionuclides on dose because any nonequilibrium in radionuclides having half-lives shorter than 180 days, produced by decay from long-lived parent radionuclides, will not affect the average annual dose to the reasonably maximally exposed individual. The potential removal of radionuclides that are incorporated into crops, which could then be harvested from the fields and gardens, is not considered by DOE. Radionuclides incorporated in these plants and animal waste are assumed to be returned to the soil as fertilizer. Because crops grown in Amargosa Valley are assumed by DOE to be consumed by the reasonably maximally exposed individual or local livestock, the NRC staff notes that the modeling assumption to neglect radionuclide removal in crops and include radionuclide return to soil does not underestimate the dose. For the reasons described previously, the NRC staff notes that DOE's assumptions for radionuclide return to soil and removal in the surface soil submodel are reasonable.

Data Sufficiency and Data Uncertainty

The NRC staff evaluated data sufficiency and uncertainty for the irrigation rate source term. The irrigation rates were determined separately for field and garden crops. Irrigation rates directly affect radionuclide concentrations in the soil, because more radionuclide mass is added to the soil when the irrigation rate is higher. An average irrigation rate was calculated from irrigation rates from several crops on the basis of current practices at Amargosa Valley, Nevada. Vegetables and fruit were assumed to be grown in gardens, whereas grains and forage were assumed to be grown in fields. An average irrigation rate was used due to crop rotation in fields and gardens in Amargosa Valley. DOE accounted for crop overwatering to prevent the buildup of soluble salts in the rooting zone. Overwatering introduces more contaminated groundwater than is needed to grow the crops. The NRC staff views overwatering to be a standard approach for determining the irrigation rate for crops, because limiting salt buildup in soils is desired and practiced worldwide [Hillel, p. 229 (1971aa)]. The NRC staff notes that the DOE assumptions for irrigation source term in the surface soil submodel do not underestimate the potential dose and are consistent with present knowledge of the Yucca Mountain region and consistent with semiarid conditions.

The NRC staff evaluated data sufficiency and uncertainty for surface soil submodel parameters. DOE developed parameters for the surface soil submodel from surveys of land use in Amargosa Valley (e.g., type of crops grown, crop rotation, and crop acreage). DOE applied documented physical and chemical properties (e.g., soil properties, radionuclide properties/characteristics). DOE generally field checked or verified the data obtained from the surveys against independent records. The data were also typically collected over several years to account for variability. Documented physical and chemical parameters were obtained from measurements and analyses by independent groups, such as the U.S. Department of Agriculture's soil surveys and established literature sources. DOE determined parameter values using relationships between parameters and measured quantities, which have been published in the scientific literature (e.g., Food and Agriculture Organization and the U.S. Department of Agriculture Natural Resources Conservation Service), and documented its analyses in the Biosphere Model Report (SNL, 2007ac). DOE used parameter distributions to account for parameter uncertainty. The NRC staff reviewed the parameter distributions and notes that these distributions are representative of the range of conditions in Amargosa Valley. Parameters were also adjusted by DOE to reflect differing climate conditions, where appropriate. For example, the average irrigation rate was adjusted to represent projected future climates. The NRC staff notes that adjusting the surface soil submodel parameters to account for climate changes, on the basis of cautious and reasonable assumptions, is consistent with the characteristics of the reference biosphere. Because DOE addressed parameter uncertainty for differing climate conditions in a similar manner to the representation of the present-day climate (discussed previously), the NRC staff notes that the resulting parameter distributions address uncertainty.

Model Uncertainty and Model Support

The NRC staff evaluated DOE's model support and treatment of model uncertainty for the surface soil submodel. In SNL Section 6.3.3 (2007ac), DOE found that there are no alternative conceptual models for the biosphere evaluation. For its review of the redistribution of radionuclides in soil, the NRC staff evaluated this conclusion in terms of the surface soil submodel. Because the DOE surface soil model relies on first principles of mass balance to represent radionuclide redistribution in soil, the NRC staff notes that this DOE result is reasonable. The NRC staff is not aware of an alternative approach to representing radionuclide redistribution in soil that uses a conceptual model that is significantly different from the first-principles approach used in the DOE surface soil model. DOE compared ERMYN with two other established models that assess radionuclide concentrations in soil, GENII (Napier, et al., 2006aa) and BIOMASS ERB2A (International Atomic Energy Agency, 2003aa), to evaluate the technique used to solve the mathematical model. The mathematical development all these models used, including the surface soil submodel used in ERMYN, is similar after the terms are combined or redefined, as identified in SNL Section 7.3.1.1 (2007ac). DOE explained differences in the models and concluded that the calculations were equivalent. The NRC staff notes that the differences were explained, and differences in these models are not expected to significantly affect performance assessment results. An external review conducted by independent experts [SNL Section 7.6 (2007ac)] provides additional confidence in the ERMYN model. This external review covered a broader scope including surface soil components; it did not explicitly address the surface soil submodel. Nonetheless, the external review concluded in SNL Section 7.6 (2007ac) that the overall ERMYN model was a well-constructed, transparent, and complete biosphere modeling tool. On the basis of its review of the comparisons to established models, the NRC staff notes that DOE's assessment of model uncertainty and model support for the surface soil submodel is reasonable.

NRC Staff Evaluation of the Groundwater Exposure Scenario

On the basis of the previously described evaluation, the NRC staff notes that the surface soil submodel provides a reasonable approach for calculating the radionuclide concentrations in both cultivated field and garden surface soils following radionuclide release in the groundwater pathway. DOE integrated its surface soil submodel by incorporating the important processes associated with the three included features, events, and processes. DOE described the governing processes of radionuclide buildup, retention, and removal in the surface soil. The NRC staff notes that the parameter determinations and their uncertainties for the surface soil submodel are described and justified. The NRC staff notes that the data support the abstracted model for radionuclide transport in the soil because they are based on documented soil properties of the Yucca Mountain region. The NRC staff notes that data uncertainty was characterized and propagated through the abstracted model by stochastic sampling of parameter ranges. The NRC staff notes that DOE addressed uncertainty in the conceptual model through comparisons to soil submodels in two other established biosphere models. Because DOE supported the surface soil model results by comparison to results from other published biosphere models, DOE's model support is reasonable.

The NRC staff has reviewed information in the SAR and other supporting information. DOE has provided information that addresses the criteria described in YMRP Section 2.2.1.3.13.3 for the surface soil submodel and its implementation in the GoldSim modeling environment of the TSPA analysis.

2.2.1.3.13.4 NRC Staff Conclusions

NRC staff notes that the DOE description of this model abstraction for airborne transport and redistribution of radionuclides is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.13.5 References

Andronico, D., S. Scollo, S. Caruso, and A. Cristaldi. 2008aa. "The 2002–03 Etna Explosive Activity: Tephra Dispersal and Features of the Deposits. *Journal of Geophysical Research*. Vol. 113, No. B04209. doi:10.1029/2007JB005126.

Anspaugh, L.R., S.L. Simon, K.I. Gordeev, I.A. Likhtarev, R.M. Maxwell, and S.M. Shinkarev. 2002aa. "Movement of Radionuclides in Terrestrial Ecosystems by Physical Processes." *Health Physics.* Vol. 82, No. 5. pp. 669–679.

Benke, R.R., D.M. Hooper, J.S. Durham, D.R. Bannon, K.L. Compton, M. Necsoiu, and R.N. McGinnis, Jr. 2009aa. “Measurement of Airborne Particle Concentrations Near the Sunset Crater Volcano, Arizona.” *Health Physics*. Vol. 96, No. 2. pp. 97–117.

Bird, R.B., W.E. Stewart, and E.N. Lightfoot. 1960aa. *Transport Phenomena*. New York City, New York: John Wiley & Sons. pp. 74-76.

Blong, R.J. 1984aa. *Volcanic Hazards—A Sourcebook on the Effects of Eruptions*. Orlando, Florida: Academic Press. P. 427.

BSC. 2006ad. “Inhalation Exposure Input Parameters for the Biosphere Model.” ANL–MGR–MD–000001. Rev. 04. ACN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2006ah. “Soil-Related Input Parameters for the Biosphere Model.” ANL–NBS–MD–000009. Rev. 03. AD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2003ad. “Characterize Eruptive Processes at Yucca Mountain, Nevada.” ANL–MGR–GS–000002. Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

CNWRA. 2007aa. “Software Validation Report for Total-System Performance Assessment (TPA) Code Version 5.1.” ML072840502. San Antonio, Texas: CNWRA.

Codell, R.B. 2004aa. “Alternate Igneous Source Term Model for Tephra Dispersal at the Yucca Mountain Repository.” *Nuclear Technology*. Vol. 148. pp. 205–212.

Codell, R.B. 2003aa. “Alternative Igneous Source Term Model for the Yucca Mountain Repository.” Proceedings of the 10th International High-Level Radioactive Waste Management Conference (IHLRWM), Las Vegas, Nevada, March 30–April 2, 2003. LaGrange Park, Illinois: American Nuclear Society.

CRWMS M&O. 1997aa. “Regional and Local Wind Patterns Near Yucca Mountain.” B00000000–01717–5705–00081. Rev. 00. MOL.19980204.0319. Las Vegas, Nevada: CRWMS M&O.

DOE. 2009ab. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 2.” Letter (February 23) J.R. Williams to J.H. Sulima (NRC). ML090550101. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009av. DOE/RW–0573, “Safety Analysis Report Yucca Mountain Repository License Application.” Rev. 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bk. “Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.10 and 2.3.11), Safety Evaluation Report Vol. 3, Chapter 2.2.1.13, Set 1.” Letter (July 27) J.R. Williams to J.H. Sulima (NRC). ML092090273. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bl. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.1, 2.3.11, 2.4.3, and 2.4.4), Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML092110472. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009bm. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.10 and 2.3.11), Safety Evaluation Report Vol. 3, Chapter 2.2.1.13, Set 1." Letter (September 17) J.R. Williams to J.H. Sulima (NRC). ML092610250. Las Vegas Nevada: DOE, Office of Civilian Radioactive Waste Management.

Hawkes, H.E. 1976aa. "The Downstream Dilution of Stream Sediment Anomalies." *Journal of Geochemical Exploration*. Vol. 6. pp. 345−358.

Heffter, J.L. and B.J.B. Stunder. 1993aa. "Volcanic Ash Forecast Transport and Dispersion (VAFTAD) Model." *Weather and Forecasting*. Vol. 8. pp. 533–541.

Heiken, G. 1978aa. "Characteristics of Tephra from Cinder Cone, Lassen Volcanic National Park, California." *Bulletin Volcanologique*. Vol. 41, No. 2. pp. 119-130.

Hill, B.E. and C.B. Connor. 2000aa. "Technical Basis for Resolution of the Igneous Activity Key Technical Issue." ML011930254 San Antonio, Texas: CNWRA.

Hill, B.E., C.B. Connor, J. Weldy, and N. Franklin. 2000ab. "Methods for Quantifying Hazards from Basaltic Tephra-Fall Eruptions." Proceedings of the Cities on Volcanoes 2 Conference, Auckland, New Zealand, February 12–14, 2001. C. Stewart, ed. Lower Hutt, New Zealand: Institute of Geological and Nuclear Sciences Limited. Institute of Geological and Nuclear Sciences Information Series. Vol. 49. p. 158.

Hill, B.E., C.B. Connor, M.S. Jarzemba, P.C. La Femina, M. Navarro, and W. Strauch. 1998aa. "1995 Eruptions of Cerro Negro Volcano, Nicaragua and Risk Assessment for Future Eruptions." *Geological Society of America Bulletin*. Vol. 110, No. 10. pp. 1,231–1,241.

Hillel, D. 1971aa. *Soil and Water: Physical Principles and Processes*. New York City, New York: Academic Press.

Hurst, A.W. and R. Turner. 1999aa. "Performance of the Program ASHFALL for Forecasting Ashfall During the 1995 and 1996 Eruptions of Ruapehu Volcano." *New Zealand Journal of Geology and Geophysics*. Vol. 42. pp. 615–622.

Inbar, M., J. Lugo, and L. Villers. 1994aa. "The Geomorphological Evolution of the Paricutin Cone and Lava Flows, Mexico, 1943–1990." *Geomorphology*. Vol. 9. pp. 57–76.

International Atomic Energy Agency. 2003aa. "Reference Biospheres" for Solid Radioactive Waste Disposal, Report of BIOMASS Theme 1 of the BIOsphere Modelling and ASSessment (BIOMASS) Programme, Part of the IAEA Co-ordinated Research Project on Biosphere Modelling and Assessment (BIOMASS)." IAEA–BIOMASS–6. Vienna, Austria: International Atomic Energy Agency, Waste Safety Section.

Janetzke, R., R. Benke, and D. Hooper. 2008aa. “Software Validation Test Plan and Report for the Volcanic Ash Dispersal and Deposition Code TEPHRA, Version 1.0.” ML081210677 San Antonio, Texas: CNWRA.

Jarzemba, M.S. 1997aa. “Stochastic Radionuclide Distributions After a Basaltic Eruption for Performance Assessments at Yucca Mountain.” *Nuclear Technology*. Vol. 118. pp. 132–141.

Jarzemba, M.S., P.A. LaPlante, and K.J. Poor. 1997aa. “ASHPLUME Version 1.0—A Code for Contaminated Ash Dispersal and Deposition.” CNWRA 97-004. San Antonio, Texas: CNWRA.

Leopold, L.B., W.W. Emmett, and R.M. Myrick. 1966aa. “Channel and Hillslope Processes in a Semiarid Area, New Mexico.” U.S. Geological Survey Professional Paper 352–G. pp. 193–253.

Leopold, L.B., M.G. Wolman, and J.P. Miller. 1964aa. “Fluvial Processes in Geomorphology.” San Francisco, California: W.H. Freeman and Co. 1964. Reprinted in 1995. New York City, New York: Dover Publications.

Leslie, B., C. Grossman, and J. Durham. 2007aa. “Total-System Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide.” Rev. 1. ML080510329. San Antonio, Texas: CNWRA.

Marcus, W.A. 1987aa. “Copper Dispersion in Ephemeral Stream Sediments.” *Earth Surface Processes and Landforms*. Vol. 12. pp. 217−228.

Napier, B.A., D.L. Strenge, J.V. Ramsdell, Jr., P.W. Eslinger, and C. Fosmire. 2006aa. “GENII Version 2 Software Design Document.” PNNL–14584. Rev. 1. Richland, Washington: Pacific Northwest National Laboratory.

NOAA. 2004aa. “Upper Air Data for Desert Rock, Nevada, Years 1978–2003.” National Climatic Data Center Digital Upper Air Files TD 6201 and 6301. Asheville, North Carolina: National Oceanic and Atmospheric Administration.

NRC. 2009ab. "Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63." Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, “Integrated Issue Resolution Status Report.” Rev. 1. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, “Yucca Mountain Review Plan—Final Report.” Rev. 2. Washington, DC: NRC.

Pelletier, J.D. and J.P. Cook. 2005aa. “Deposition of Playa Windblown Dust Over Geologic Time Scales.” *Geology*. Vol. 33, No. 11. pp. 909–912.

Pelletier, J.D., S.B. DeLong, M.L. Cline, C.D. Harrington, and G.N. Keating. 2008aa. “Dispersion of Channel-Sediment Contaminants in Distributary Fluvial Systems: Application to Fluvial Tephra and Radionuclide Redistribution Following a Potential Volcanic Eruption at Yucca Mountain.” *Geomorphology*. Vol. 94. pp. 226–246.

Pelletier, J.D., C.D. Harrington, J.W. Whitney, M. Cline, S.B. DeLong, G. Keating, and K.T. Ebert. 2005aa. "Geomorphic Control of Radionuclide Diffusion in Desert Soils." *Geophysical Research Letters.* Vol. 32. p. L23401.

Pioli, L., E. Erlund, E. Johnson, K. Cashman, P. Wallace, M. Rosi, and H. Delgado Granados. 2008aa. "Explosive Dynamics of Violent Strombolian Eruptions: The Eruption of Parícutin Volcano 1943–1952 (Mexico)." *Earth and Planetary Science Letters*. Vol. 271. pp. 359–368.

Segerstrom, K. 1966aa. "Parícutin, 1965—Aftermath of Eruption." U.S. Geological Survey Professional Paper 550–C. pp. C93–C101.

Segerstrom, K. 1950aa. "Erosion Studies at Parícutin, State of Michoacán, Mexico." U.S. Geological Survey Bulletin 965–A. pp. 1–164.

SNL. 2008ab. "Features, Events, and Processes for the Total System Performance Assessment: Analyses." ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ab. "Atmospheric Dispersal and Deposition of Tephra From a Potential Volcanic Eruption at Yucca Mountain, Nevada." MDL–MGR–GS–000002. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ac. "Biosphere Model Report." MDL–MGR–MD–000001. Rev. 02. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ae. "Characterize Eruptive Processes at Yucca Mountain, Nevada." ANL–MGR–GS–000002. Rev. 03. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007av. "Redistribution of Tephra and Waste by Geomorphic Processes Following a Potential Volcanic Eruption at Yucca Mountain, Nevada." MDL–MGR–GS–000006. Rev. 00. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

Sparks, R.S.J., M.I. Bursik, S.N. Carey, J.S. Gilbert, L.S. Glaze, H. Sigurdsson, and A.W. Woods. 1997aa. *Volcanic Plumes*. New York City, New York: John Wiley & Sons. p. 574.

Suzuki, T. 1983aa. "A Theoretical Model for the Dispersion of Tephra." *Arc Volcanism: Physics and Tectonics*. D. Shimozuru and I. Yokoyama, eds. Tokyo, Japan: Terra Scientific Publishing Company. pp. 95–113.

Till, J.E. and R.E. Moore. 1988aa. "A Pathway Analysis Approach for Determining Acceptable Levels of Contamination of Radionuclides in Soil." *Health Physics*. Vol. 55, No. 3. pp. 541–548.

Winfrey, B. 2005aa. "Software Validation Test Plan and Report, TEPHRA, Version 1.0." ML063050217 San Antonio, Texas: CNWRA.

(Intentionally Left Blank)

CHAPTER 16

2.2.1.3.14 Biosphere Characteristics

2.2.1.3.14.1 Introduction

This chapter of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) postclosure performance assessment model used to calculate biosphere transport and the annual dose to the reasonably maximally exposed individual, as presented in DOE's Safety Analysis Report (SAR) (DOE, 2008ab). The sources of radionuclides used in DOE's biosphere model calculations are calculated by other models in its performance assessment analysis. Those models calculate repository releases from postclosure engineered-barrier-system failures and then model the transport of the released radionuclides from the repository location to the biosphere. Results from these other transport models provide the sources of radionuclides from two primary biosphere media: groundwater and soil contaminated with tephra deposits. In the model, tephra (hereafter, volcanic ash) is deposited on the ground from postulated volcanic events. DOE's biosphere model then calculates the subsequent transport of these radionuclides within the biosphere through a variety of exposure pathways (e.g., soil, food, water, air) and applies dosimetry modeling to convert the reasonably maximally exposed individual exposures into annual dose.

10 CFR 63.2 defines the reference biosphere as "the description of the environment inhabited by the [reasonably maximally exposed individual]." The reasonably maximally exposed individual is defined at 10 CFR 63.312 as a hypothetical adult who (i) lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination; (ii) has a diet and living style representative of current Amargosa Valley, Nevada, residents; (iii) uses well water with average concentrations of radionuclides based on an annual water demand of 3.7×10^9 L [3,000 acre-ft]; (iv) drinks 2 L [0.528 gal] of water per day from groundwater extracted from wells drilled at the location specified in (i) of this paragraph; and (v) is an adult with metabolic and physiological considerations consistent with present knowledge of adults.

DOE estimated the dose to the reasonably maximally exposed individual on the basis of the concentrations of radionuclides in groundwater and in contaminated ash. These concentrations were calculated by DOE's model abstractions for saturated zone transport [Safety Analysis Report (SAR) Revision 1 Section 2.3.9], extrusive (volcanic eruption) atmospheric dispersal (SAR Section 2.3.11.4.5.2), and volcanic ash redistribution (SAR Section 2.3.11.4.5.3). These model abstractions, which provide inputs for the biosphere calculations within the Total System Performance Assessment (TSPA) model, are reviewed in Technical Evaluation Report (TER) Sections 2.2.1.3.9 and 2.2.1.3.13, respectively. This chapter of the TER focuses on the NRC staff's review of the performance assessment calculations of biosphere transport and dose to the reasonably maximally exposed individual described in SAR Section 2.3.10. The NRC staff's evaluation of biosphere modeling of radionuclide concentrations in soil can be found in TER Section 2.2.1.3.13.

In SAR Section 2.3.10, DOE analyzed the characteristics of the Yucca Mountain region and Amargosa Valley, Nevada, for its biosphere transport and reasonably maximally exposed individual dose calculations. DOE identified features, events, and processes (FEPs) and developed biosphere conceptual and mathematical models for use in its TSPA computer model.

DOE described environmental conditions, resident lifestyle, exposure media, environmental transport pathways, and human exposure pathways it used for evaluating the impacts of repository performance on dose to the reasonably maximally exposed individual. Exposure pathways in the DOE biosphere model are based on assumptions about residential and agricultural uses of the water and indoor and outdoor activities. These pathways include ingestion, inhalation, and direct exposure to radionuclides deposited to soil from irrigation (SAR Section 2.3.10.1). Ingestion pathways include drinking contaminated water, eating crops irrigated with contaminated water, eating food products produced from livestock raised on contaminated feed and water, eating farmed fish raised in contaminated water, and inadvertently ingesting soil. Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products.

DOE's approach to biosphere modeling was twofold. DOE used a standalone computer code entitled Environmental Radiation Model for Yucca Mountain Nevada (ERMYN) to calculate biosphere dose conversion factors, which were used as inputs in DOE's TSPA code. The TSPA multiplied the appropriate biosphere dose conversion factor by either a soil concentration or a water concentration to obtain the dose to the reasonably maximally exposed individual for each exposure scenario (i.e., volcanic ash, groundwater). The substance of DOE's biosphere modeling approach is contained primarily in the ERMYN code.

This chapter evaluates the technical bases for DOE's conceptual and mathematical biosphere models, input parameter selections, parameter uncertainty propagation, model support, and model implementation and integration within DOE's performance assessment evaluation. These evaluations are organized by subsections that address specific components of DOE's biosphere model (or model development process), including system description and model integration, biosphere transport pathways, human exposure, dosimetry, and integrated biosphere modeling results. The NRC staff's review evaluates both the biosphere modeling in the ERMYN code and how DOE used the ERMYN output (the biosphere dose conversion factors) to calculate reasonably maximally exposed individual dose in the TSPA model.

2.2.1.3.14.2 Evaluation Criteria

NRC staff's review of model abstractions used in the DOE postclosure performance assessment, including those considered in this chapter for climate and infiltration, is guided by 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments). The DOE Total System Performance Assessment is reviewed in TER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require that a performance assessment

- Include appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is in TER Chapter 2.2.1.2.1. 10 CFR 63.114(a) provides requirements for performance assessment for the initial 10,000 years following disposal. 10 CFR 63.114(b) and 63.342 provide requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections require that through the period of geologic stability, with specific limitations, the DOE dose calculation should

- Use performance assessment methods consistent with the performance assessment methods used to calculate dose for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

10 CFR 63.305 states the following requirements for characteristics of the reference biosphere:

- Features, events, and processes that describe the reference biosphere are consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site.

- Changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology should not be projected; in all analyses, all of those factors should be assumed to be constant as they are at the present time.

- Factors related to the geology, hydrology, and climate should be varied based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, consistent with the criteria for performance assessments specified at 10 CFR 63.342.

- Biosphere pathways are consistent with arid or semiarid conditions.

10 CFR 63.312 states the following requirements for characteristics of the reasonably maximally exposed individual (RMEI):

- Lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination

- Has a diet and living style representative of the people who now reside in the Town of Amargosa Valley, Nevada (projections are based upon surveys of the people residing in the Town of Amargosa Valley, Nevada, to determine their current diets and living styles); mean values of these factors are to be used in performance assessment

- Uses well water with average concentrations of radionuclides based on an annual water demand of 3000 acre-ft

- Drinks 2 L [0.07 ft^3] of water per day from wells drilled into the groundwater in the accessible environment where the RMEI resides

- Is an adult with metabolic and physiological considerations consistent with present knowledge of adults

- Has biosphere pathways consistent with arid or semi-arid conditions

The approach to be taken for implementation of the total effective dose equivalent (TEDE) is specified at 10 CFR 63.102 (Concepts).

The NRC staff review of the SAR and supporting information follows the guidance laid out in YMRP Section 2.2.1.3.14, Biosphere Characteristics, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff evaluation of DOE's abstraction of biosphere characteristics are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

NRC staff review used a risk-informed approach and the guidance provided by the YMRP, as supplemented by NRC (2009ab), to the extent reasonable for aspects of biosphere characteristics important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this chapter. The NRC staff's determination was based both on risk information provided by DOE and on NRC staff knowledge gained through experience and independent confirmatory analyses.

2.2.1.3.14.3 Technical Evaluation

The NRC staff's technical review of DOE's biosphere characteristics model abstraction evaluated both the biosphere model and the model development process. The review focused on five topics: (i) system description and model integration, (ii) biosphere transport pathways, (iii) human exposure, (iv) dosimetry, and (v) the integrated biosphere modeling results. These reviews are documented in subsections of this TER chapter. The system description and model integration review evaluated DOE's overall conceptualization of the biosphere including features, events, and processes that were selected and included in DOE's biosphere conceptual models.

The NRC staff's detailed review focused on the most risk-significant parts of DOE's biosphere model. Risk insights that apply to both DOE's TSPA results and to DOE's detailed abstraction modeling of the biosphere (i.e., using the ERMYN code to generate the biosphere dose conversion factors) informed the NRC staff's review. These risk insights focused the NRC staff's detailed review on those aspects of DOE's biosphere modeling that contributed most to the calculated reasonably maximally exposed individual dose results in the TSPA.

The NRC staff analyzed the risk-significance aspects of the TSPA biosphere model abstraction in the TSPA code by evaluating DOE's sensitivity analysis results using the TSPA code. These results indicated that biosphere dose conversion factors (ERMYN code outputs) significantly affected the TSPA results, as identified in SNL, Appendix K, pp. FK–63 to FK–65 (2008ag). The NRC staff therefore performed a detailed technical evaluation of DOE's biosphere model.

The NRC staff's risk-informed review evaluated a subset of DOE's biosphere model abstraction to determine whether DOE's overall methodology was reasonable. This detailed review focused on the subset of radionuclides and exposure pathways that were the most risk significant in DOE's performance assessment analysis. These radionuclides and exposure pathways are summarized in TER Tables 16-1 and 16-2.

On the basis of DOE's documentation of its performance assessment results, the NRC staff developed TER Tables 16-1 and 16-2 using the following two-step approach. First, the NRC staff identified those radionuclides that individually account for the largest fraction of DOE's peak total mean annual dose results from its TSPA analysis, as shown in SAR Figure 2.4-20. Next, the primary pathways that account for the largest fraction of DOE's calculated biosphere dose conversion factors for each identified radionuclide were included. DOE's pathway contributions to each radionuclide-specific biosphere dose conversion factor were documented in SAR Table 2.3.10-11. On the basis of this analysis of DOE's results, the NRC staff notes that the radionuclides and pathways identified in TER Tables 16-1 and 16-2 are the most risk-significant contributors to DOE's TSPA results.

The NRC staff focused its detailed review on the subset of radionuclides and pathways that are the most risk-significant contributors to DOE's performance assessment results. An example is provided here to clarify how this approach identifies the most risk-significant contributors to DOE's results. For DOE's 1-million-year results presented in SAR Figure 2.4-20(b), the peak

Table 16-1. Exposure Pathways and Radionuclides Determined the Most Risk Significant in the DOE Performance Assessment for the 10,000-Year Simulation Period

Radionuclide*	Source of Radionuclides†	Route of Exposure‡	Primary Pathways‡
Tc-99	Estimated Releases to Groundwater	Ingestion	42% drinking water
			37% animal product
C-14		Ingestion	59% fish
			22% drinking water
Pu-239		Inhalation	50% particulates
			24% evaporative cooler aerosols
		Ingestion	19% drinking water
I-129		Ingestion	60% drinking water
			28% animal products

*Radionuclides presented in order of their contribution to the DOE peak total mean annual dose results in SAR Figure 2.4-20.
†Modeling cases that contribute most to the DOE total mean annual dose are based on release to groundwater as shown in SAR Figure 2.4-18 and SAR Section 2.4.2.2.1.2.
‡Routes of exposure and primary pathways from SAR Table 2.3.10-11. Various pathways not listed contribute the remaining percentage of each radionuclide dose.

Table 16-2. Exposure Pathways and Radionuclides Determined the Most Risk Significant in the DOE Performance Assessment for the 1-Million-Year Simulation Period

Radionuclide*	Source of Radionuclides†	Route of Exposure‡	Primary Pathways‡
Pu-242	Estimated Releases to Groundwater	Inhalation	51% particulates
			24% evaporative cooler aerosols
		Ingestion	19% drinking water
Np-237		Inhalation	35% evaporative cooler aerosols
			21% particulates
		Ingestion	29% drinking water
Ra-226		Inhalation	74% radon
I-129		Ingestion	60% drinking water
			28% animal products

*Radionuclides presented in order of their contribution to the DOE peak total mean annual dose results in SAR Figure 2.4-20.
†Modeling cases that contribute most to the DOE total mean annual dose are based on release to groundwater as shown in SAR Figure 2.4-18 and SAR Section 2.4.2.2.1.2.
‡Routes of exposure and primary pathways from SAR Table 2.3.10-11. Various pathways not listed contribute the remaining percentage of each radionuclide dose.

total mean annual dose is approximately 0.02 mSv/yr [2 mrem/yr]; four radionuclides contribute approximately 0.015 mSv/yr [1.5 mrem/yr] (75 percent) to that value. The NRC staff identified these four radionuclides as the most risk-significant contributors because they represent the smallest number of radionuclides that comprise the largest fraction of the peak mean dose. The remaining 17 radionuclides in DOE's analysis each contributed a small fraction to the peak mean dose. The pathways for these four radionuclides were then individually evaluated to identify the subset of pathways that contributed the largest fraction to the dose contribution from the radionuclide using information provided in SAR Table 2.3.10-11. For example, Pu-242 is responsible for 30 percent of DOE's peak mean dose. The NRC staff evaluated the pathways through which Pu-242 contributed to the dose and noted that the inhalation of particulates pathway was responsible for 51 percent of the Pu-242 dose, the inhalation of evaporative cooler aerosols pathway was responsible for 24 percent of the Pu-242 dose, and the drinking groundwater pathway was responsible for 19 percent of the Pu-242 dose. Twelve other pathways are responsible for the remaining 6 percent of the Pu-242 dose; therefore, three pathways were identified as being the most risk significant for Pu-242. This example illustrates the NRC staff's approach to identifying the radionuclides and their pathways that are the most risk significant to DOE's performance assessment calculation.

TER Table 16-1 contains the radionuclides and their pathways that are the most risk -significant contributors to DOE's performance assessment dose results for the time period of 10,000 years following disposal. TER Table 16-2 contains the radionuclides and their pathways that are the most risk-significant contributors to DOE's performance assessment dose results for the time period after the initial 10,000 years up to 1 million years. The radionuclides listed in TER Tables 16-1 and 16-2 include radionuclides determined to be important contributors to dose results in prior independent NRC performance assessment results, as identified in NRC Volume 2, Appendix D (2005aa).

While the NRC staff's technical review evaluated all of DOE's biosphere modeling documentation at a general level of detail, the NRC staff's focused review evaluated DOE's biosphere submodels and input parameters that are risk significant in the biosphere dose conversion factor calculations on the basis of the NRC staff's identification of risk-significant pathways in TER Tables 16-1 and 16-2. In particular, the NRC staff's review of the data supporting the biosphere transport pathway input parameters (TER Section 2.2.1.3.14.3.2) focused on parameters in DOE's transport submodels. These transport submodels include plant uptake, animal uptake, fish uptake, and air modeling. Similarly, the NRC staff's review of DOE's data supporting input parameters for the human exposure submodels evaluated in TER Section 2.2.1.3.14.3.3 focused on the inhalation and ingestion exposure submodels because those are the routes of exposure identified to be most risk significant in Tables 16-1 and 16-2.

2.2.1.3.14.3.1 System Description and Model Integration

In SAR Section 2.3.10.2 and in supporting references, DOE described the biosphere characteristics of the Yucca Mountain region; of Amargosa Valley, Nevada, that impact its residents; of included features, events, and processes; and of the biosphere conceptual models in the ERMYN code that were used to calculate biosphere dose conversion factors. This section documents the NRC staff's review of these descriptions. An additional part of the NRC staff's review evaluated integration (i.e., couplings, consistency, and assumptions) of the TSPA biosphere model abstraction with other TSPA model abstractions.

Features, Events, and Processes

DOE described the Yucca Mountain region characteristics in SAR Section 2.3.10.2.1. This information addressed topics including climate, topography and soils, native flora and fauna (i.e., plants and animals), local communities, infrastructure (including water source), and agricultural conditions. Information on the characteristics of Amargosa Valley, Nevada, residents (summarized in SAR Section 2.3.10.2.2) originated predominantly from local and national surveys. The SAR addressed topics such as diet and lifestyle factors, including the use of evaporative coolers, gardening, employment, commuting, housing, and metabolic considerations. DOE documented the screening approach for the features, events, and processes in SAR Section 2.2.1.2 and listed all the features, events, and processes that were evaluated for the TSPA model in SAR Table 2.2-1. Features, events, and processes that were included in the biosphere model were tabulated in SAR Table 2.3.10-1 and are reviewed in this TER section. The NRC staff's review of excluded features, events, and processes is documented in TER Section 2.2.1.2.1.3.2.

The NRC staff's review evaluated the technical bases DOE used to support its disposition of included features, events, and processes in the performance assessment with respect to the following: (i) whether DOE provided reasonable technical bases for including biosphere features, events, and processes; (ii) whether the included features, events, and processes were consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site; and (iii) whether DOE included all biosphere-related features, events, and processes that could significantly change the magnitude or timing of the radiological exposures to the reasonably maximally exposed individual in its performance assessment.

The NRC staff evaluated DOE's technical bases for included features, events, and processes and reviewed DOE's descriptions of how each included biosphere feature, event, and process was incorporated into the performance assessment calculation. In this review, the NRC staff verified that the features, events, and processes which could significantly contribute to the reasonably maximally exposed individual dose were included in the performance assessment calculation and that information supporting the features, events, and processes was based on present knowledge of the Yucca Mountain region conditions. Because many features, events, and processes are general in nature (e.g., climate change, biosphere characteristics, plant uptake), the NRC staff evaluated whether the performance assessment evaluation included the specific aspects of a feature, event, and process that would be expected to contribute to reasonably maximally exposed individual dose and therefore should be included in the model. As part of its review of DOE's analyses, the NRC staff also incorporated its understanding of Yucca Mountain region characteristics obtained from prior experience and independent analyses of the biosphere characteristics at Yucca Mountain (e.g., LaPlante and Poor, 1997aa).

The NRC staff notes that DOE's list of included features, events, and processes is consistent with the NRC staff's independent assessment of the characteristics of the Yucca Mountain region and Amargosa Valley. The NRC staff's review noted that features included in DOE's performance assessment (i.e., wells, soil type, agricultural land use, irrigation, animal farms, fisheries, and human lifestyle and characteristics) were supported by the technical bases in the SAR and supporting references and are representative of the present knowledge of the Yucca Mountain region. The NRC staff's review noted that processes included in DOE's performance assessment (i.e., radionuclide accumulation in soils, atmospheric transport, plant and animal uptake, radioactive decay, ingestion, inhalation, external exposure, and radiation dose) were supported by the technical bases in the SAR and supporting references that represented present knowledge of the Yucca Mountain region. The NRC staff notes that DOE's list of included features, events, and processes is suitable for use in its biosphere model. In addition, the NRC staff has not identified any additional features, events, and processes that are excluded from DOE's biosphere model that would be expected to significantly increase the dose or affect the timing of dose to the reasonably maximally exposed individual.

Conceptual Models

In SAR Section 2.3.10.2.3, DOE considered features, events, and processes for the reference biosphere model and dose calculation of the reasonably maximally exposed individual in identifying applicable exposure pathways and developing exposure scenarios for modeling dose to the reasonably maximally exposed individual. An exposure scenario, in general, describes a set of facts, assumptions, and inferences about how exposure occurs. In DOE's Yucca Mountain biosphere model, an exposure scenario is a conceptual model that describes the biosphere characteristics which lead to the reasonably maximally exposed individual's exposure to radionuclides that enter the biosphere from different transport routes (i.e., groundwater or volcanic ash). DOE's conceptual representations of the exposure pathways for groundwater and volcanic ash exposure scenarios were provided in SAR Figures 2.3.10-6 and 2.3.10-8. DOE incorporated these conceptual representations into mathematical submodels in the ERMYN code. The mathematical submodels in the ERMYN code were depicted in SAR Figures 2.3.10-9 and 2.3.10-10 and described in SAR Sections 2.3.10.2.5 and 2.3.10.2.6.

The NRC staff evaluated DOE's conceptual representations (i.e., conceptual models) and associated mathematical submodels in the ERMYN code. The NRC staff reviewed both DOE's groundwater exposure scenario (the modeling of biosphere characteristics that lead to the reasonably maximally exposed individual exposure to radionuclides from contaminated groundwater) and the volcanic ash exposure scenario. These reviews, documented in the subsections that follow, evaluated the potential pathways of radionuclide transport and exposure in DOE's conceptual representations of the biosphere model. NRC staff review of the portions of the biosphere and reasonably maximally exposed individual criteria that are applicable to the input parameters and data is discussed in TER Sections 2.2.1.3.14.3.2, 2.2.1.3.14.3.3, and 2.2.1.3.14.3.4.

Groundwater Exposure Scenario Conceptual Model

The NRC staff's review of DOE's conceptual model of the groundwater exposure scenario included biosphere features, events, and processes, their functional relationships, and the resulting exposure pathways for modeling biosphere transport and dose to the reasonably maximally exposed individual. The NRC staff's evaluation of functional relationships among features, events, and processes considered how DOE accounted for interactions among related features, events, and processes in the biosphere conceptual model so that all potential pathways could be identified. For example, farming practices, such as soil irrigation, can lead to soil accumulation of radionuclides, which can contribute to plant uptake of radionuclides from that soil.

DOE's groundwater exposure scenario includes a reasonably maximally exposed individual who is assumed to be an adult who lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination. The reasonably maximally exposed individual is assumed to use wells to draw groundwater from the contamination plume for domestic and agricultural purposes. The NRC staff notes that the general reasonably maximally exposed individual and biosphere characteristics in DOE's model address the use of groundwater for drinking, irrigating crops, watering livestock, raising fish, and operating evaporative coolers and provide a reasonable approach.

DOE's conceptualization of dose to the reasonably maximally exposed individual involves three routes of exposure: external exposure, inhalation, and ingestion. The inhalation dose portion of DOE's conceptual model includes reasonably maximally exposed individual inhalation of radionuclides in (i) resuspended soil particles, (ii) gaseous emissions from the soil and their radioactive decay products, and (iii) aerosols generated by evaporative coolers. The ingestion dose portion of DOE's conceptual model includes (i) drinking water; (ii) crops, including leafy vegetables, other vegetables, fruits, and grains; (iii) animal products, including meat, poultry, milk, and eggs; (iv) freshwater fish; and (v) soil. The meat category is a combination of all edible portions of beef, pork, and wild game (BSC, 2005ab).

On the basis of the preceding NRC staff's review of the biosphere characteristics that are evaluated in DOE's groundwater exposure scenario conceptual model (described in SAR Section 2.3.10.2) and the associated mathematical models described in SAR Section 2.3.10.3, the NRC staff notes that these models include all potential pathways of radionuclide transport and exposure. This is based on DOE's inclusion of pathways that (i) the NRC staff considers characteristic of the Yucca Mountain region, (ii) are commonly included in dose models and assessments, and (iii) are based on unique site-specific considerations (e.g., evaporative coolers, fish farming, wild game, and radon). The NRC staff notes that DOE's groundwater conceptual model and associated mathematical models are reasonable.

Volcanic Ash Exposure Scenario Conceptual Model

The NRC staff's review of the volcanic ash exposure scenario conceptual model evaluated the included biosphere system features, events, and processes; their functional relationships; and the included exposure pathways for modeling biosphere transport and dose to the reasonably maximally exposed individual. The review noted that DOE's volcanic ash exposure scenario includes reasonably maximally exposed individual exposure to radioactive waste entrained in (i) volcanic ash that is deposited from the initial plume of released radionuclides directly to the ground at the location of the reasonably maximally exposed individual and (ii) regional volcanic ash deposits that are redistributed via eolian (carried by wind) and fluvial (carried by water) processes. Other models in DOE's performance assessment address the transport of contaminated volcanic ash to the ground in the biosphere. Those models are reviewed in TER Section 2.2.1.3.13. Because DOE calculates the reasonably maximally exposed individual dose from volcanic ash in the soil on the basis of the potential routes of exposure to the reasonably maximally exposed individual that are similar to those already reviewed for the groundwater scenario (including external, inhalation, and ingestion exposures), this review emphasizes the aspects of the volcanic exposure scenario conceptual model that are not duplicated in the groundwater exposure scenario conceptual model. DOE's sharing of submodels with the groundwater scenario is technically reasonable because the methods for calculating dose to the reasonably maximally exposed individual from radionuclide-contaminated soil are the same once the radionuclide concentration of the soil, whether from irrigation or volcanic ash, is calculated.

DOE's conceptualization of inhalation dose in the volcanic ash exposure scenario includes resuspension of radionuclides in soil particles and release of Rn-222 (radon) gas. The NRC staff notes that DOE's inclusion of resuspension modeling is consistent with DOE's description of an arid climate characterized by low rainfall (SAR Sections 2.3.10.2.1 and 2.3.1). DOE's inclusion of resuspension in its conceptual approach also addressed a variety of dust-generating activities and reasonably maximally exposed individual exposure environments (SAR Sections 2.3.10.2.6 and 2.3.10.3.2.2). Therefore, DOE has reasonably addressed the potential dust inhalation exposure pathways. DOE's ingestion dose calculations applied the same soil-contamination-based exposure pathways that were used in DOE's scenario for groundwater exposure. DOE considered that groundwater pathways which did not include a soil component (e.g., evaporative coolers, ingestion of groundwater, and ingestion of fish) did not apply to the volcanic ash exposure scenario, based upon DOE's exclusion of features, events, and processes for ash in groundwater (FEP 1.2.04.07.0B; SNL, 2008ab). The NRC staff's review of DOE's exclusion of this FEP is documented in TER Section 2.2.1.2.1.3.2.

On the basis of characteristics of DOE's volcanic ash exposure scenario conceptual model, the NRC staff notes that the conceptual model includes all potential pathways of radionuclide transport and exposure. This is based on DOE's inclusion of pathways that (i) the NRC staff considers characteristic of the Yucca Mountain region, (ii) are commonly included in dose models and assessments, and (iii) are based on unique site-specific considerations (e.g., various particulate resuspension exposure environments and radon).

Integration of Biosphere Model in the TSPA

The NRC staff reviewed the integration between the biosphere model abstraction and other TSPA abstractions for couplings among models that share or utilize similar information and consistency of assumptions among models. This review was conducted because the abstraction models are designed and implemented to function as intended within the larger TSPA model (i.e., appropriately receive and pass data), and the abstraction models that share features, events, and processes (e.g., climate change can affect both Yucca Mountain infiltration and biosphere conditions) are expected to have consistent representations of features, events, and processes (e.g., assumptions) to avoid bias in TSPA calculations. Based upon the NRC's review of DOE's integration of the TSPA biosphere model abstraction with other TSPA model abstractions, the NRC staff notes that the integration of the biosphere abstraction is reasonable for calculating the biosphere transport and dose to the reasonably maximally exposed individual in the TSPA code. This is based on the evaluation of direct couplings between the TSPA biosphere abstraction and other TSPA model abstractions and evaluation of shared assumptions in the biosphere and other abstractions.

The NRC staff's evaluation of direct couplings between the TSPA biosphere abstraction and other TSPA model abstractions considered the flow of information from other abstractions to the biosphere abstraction within selected TSPA model files. Specifically, DOE's model file for seismic ground motion for the 1-million-year modeling case passes radionuclide-specific saturated zone model results (radionuclide groundwater concentrations) to the biosphere model where they are multiplied by the groundwater exposure scenario biosphere dose conversion factors. DOE's model file for the volcanic eruption modeling case passes the ash redistribution model results (radionuclide and pathway-specific soil concentrations) for multiplication by the volcanic ash exposure scenario radionuclide and pathway-specific biosphere dose conversion factors. These evaluations show that couplings between the biosphere model abstraction and other model abstractions are in agreement with DOE's documentation of the calculations (SAR Sections 2.3.10.5.1.2 and 2.3.10.5.2.2). The NRC staff's technical expertise gained from developing and using performance assessment models confirms that these couplings are appropriate for use in DOE's performance assessment.

The NRC staff also evaluated other, less direct, points of model integration addressing consistency between the TSPA biosphere abstraction and other abstractions, including assumptions. The NRC staff noted that the biosphere model abstraction includes biosphere dose conversion factors applicable to the radionuclides identified in the inventory analysis that was developed for the postclosure performance analyses (SAR Section 2.3.7.4.1.2). Consistency in the radionuclides evaluated in the TSPA abstractions is important to ensure that the biosphere model includes biosphere dose conversion factors for the radionuclides that are included in the other TSPA abstraction models. The NRC staff also evaluated how DOE integrated climate evolution in its biosphere modeling with the climate evolution considered in the infiltration and unsaturated zone flow process models. The NRC staff evaluated DOE's technical bases for this climate implementation approach for the biosphere and noted that DOE reasonably explained that the use of current climate biosphere dose conversion factors (i.e., not explicitly modeling separate climate states in the biosphere model) is conservative and adequate for use in the reasonably maximally exposed individual dose calculations throughout the period of geologic stability in the biosphere model. DOE quantitatively evaluated the effects of climate change as follows. DOE's analysis (i) evaluated biosphere model parameters on the basis of the expected parameters impacted by climate change, (ii) derived values for these parameters on the basis of its analysis of potential future climate states, and (iii) executed biosphere calculations for three separate climate states (present-day interglacial, monsoon, glacial transition). The results showed that future climate evolution in the biosphere lowers dose to the reasonably maximally exposed individual (SAR Section 2.3.10.5.1.1). The NRC staff notes that DOE's methods, including use of the same biosphere model with different sets of climate-dependent input parameters to evaluate the effect of climate change on biosphere dose conversion factor results, are reasonable for evaluating whether TSPA calculations should include separate sets of biosphere dose conversion factors for each climate state. DOE's result was also consistent with the results from prior NRC-sponsored biosphere analyses (LaPlante and Poor, 1997aa). Both the NRC's and DOE's analyses suggested that future climate states, which are expected to be cooler and wetter than the current climate, would result in the reasonably maximally exposed individual using less water (e.g., irrigation) and therefore lower the amount of radionuclides deposited to soils and lower the calculated reasonably maximally exposed individual dose. The NRC staff evaluated whether DOE's technical basis reasonably supported its conclusion that the biosphere dose conversion factors calculated for the current climate state (modern interglacial climate) are conservative for calculating the dose to the reasonably maximally exposed individual in the TSPA biosphere abstraction throughout the period of geologic stability (SAR Section 2.3.10.5.1.1). The NRC staff notes that DOE's findings in this regard are reasonably supported.

In summary, after reviewing DOE's system description and model integration, the NRC staff notes that DOE's performance assessment is reasonable. This is based on the NRC staff's review of information discussed in this subsection, including DOE's description of the characteristics of the Yucca Mountain region; the documentation of features, events, and processes DOE has included in the biosphere model; and the integration of included features, events, and processes into the conceptual models of the biosphere system.

2.2.1.3.14.3.2 Assessment of Biosphere Transport Pathways

A series of integrated submodels in the DOE ERMYN biosphere model calculates radionuclide transport through pathways within the biosphere. Five transport submodels (surface soil, plant uptake, animal uptake, fish uptake, and air) calculate environmental media concentrations used in the ERMYN calculations of biosphere dose conversion factor input parameters for the TSPA model. The NRC staff's review of the surface soil submodel is documented in TER Section 2.2.1.3.13.3.2. This section documents the NRC staff's review of DOE's technical bases for input parameters, treatment of parameter uncertainty, and, as appropriate, evaluation of alternative conceptual models applicable to the biosphere transport submodels in ERMYN. The NRC staff's risk-informed review focused on transport submodels and applicable input parameters for exposure pathways that contribute most to the TSPA results, as discussed in TER Section 2.2.1.3.14.3.

These submodels address plant uptake, animal uptake, fish uptake, and air transport. Air transport includes localized resuspension of particulates from soil or ash, generation of indoor evaporative cooler aerosols, and the release of radon gas from soil or ash. While groundwater-release-related modeling cases are the primary contributors to the total TSPA dose results (as summarized in TER Section 2.2.1.3.14.3), the NRC's review of DOE's biosphere transport models also considered the risk-significant aspects of the volcanic-ash-related biosphere transport modeling. DOE's TSPA dose results documented in SAR Figure 2.4-32 list Pu-239 and Pu-240 as the largest contributors to its calculated peak mean annual dose for the volcanic ash modeling case. DOE further documented that reasonably maximally exposed individual inhalation of resuspended particulates was the predominant pathway for the volcanic ash exposure scenario biosphere dose conversion factors for Pu-239 and Pu-240 (SAR Table 2.3.10-15). DOE's information is consistent with the NRC staff results that show inhalation of resuspended particulates is a predominant contributor to volcanic exposure scenario dose results (NRC, 2005aa; LaPlante and Poor, 1997aa). Transport submodels for plant uptake, animal uptake, and radon are also used in volcanic ash biosphere dose conversion factor calculations, but contribute less to DOE's calculated peak mean annual dose for the volcanic ash modeling case (SAR Figure 2.4-32; SAR Table 2.3.10-15).

Plant Uptake Submodel

DOE's plant uptake submodel in the ERMYN code (SAR Section 2.3.10.3.1.3) calculates plant radionuclide concentrations on the basis of direct deposition of irrigation water and dust on plant surfaces and root uptake from estimated soil radionuclide concentrations computed by the surface soil model (or provided as direct model input for volcanic ash biosphere dose conversion factor calculations as discussed in SAR Section 2.3.10.3.2.1). For root uptake processes, soil-to-plant transfer factors are used as input parameters to calculate plant radionuclide concentration from the radionuclide concentration in the soil. DOE selected soil-to-plant transfer factors from laboratory and field study results obtained from available literature using the methods discussed in BSC (2004ap).

DOE evaluated soil-to-plant transfer factors from data obtained through a variety of references, including original data from literature reviews and biosphere analyses that selected and reported values from available sources. For its analysis, DOE identified five unique crop groups: leafy vegetables, other vegetables, fruit, grain, and forage. For each crop group and radionuclide, DOE selected soil-to-plant transfer factor values that it considered most applicable to the Yucca Mountain biosphere conditions, based upon area soil characteristics and crop types (SAR Section 2.3.10.3.1.3). DOE then calculated geometric means and standard deviations from the values selected from each reference. DOE assumed that soil-to-plant transfer factors followed truncated lognormal distributions with a 99 percent confidence interval around the geometric mean of the selected point values (BSC, 2004ap).

The NRC's review of DOE's soil-to-plant transfer factor input parameters focused on the adequacy of supporting data sources, the selection of values, the applicability of selected values to Yucca Mountain biosphere calculations, and DOE's approach to propagating uncertainty and variability in selected values. The NRC staff also reviewed the magnitude of DOE's selected geometric mean values and geometric standard deviations for key contributing radionuclides in relation to the values provided by the supporting references from which the DOE values were derived. Data sources DOE used to derive input parameters included (i) commonly referenced original data compilations that evaluated a variety of peer-reviewed field and laboratory studies and (ii) other technical analyses that reported soil-to-plant transfer factors selected or derived from available source data or compilations. These references provide a technical basis for selecting composite transfer factor values because they include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa), as well as a variety of technical reports authored by various radiological assessment practitioners, including NRC. Therefore, the NRC staff notes that the referenced documents are representative of available scientific data on soil-to-plant transfer factors, because experts in the field have reviewed and compiled the available scientific data.

A number of approaches can be used to select soil-to-plant transfer factors from crop-specific values to derive representative values for plant groups from available data sources. DOE averaged applicable point estimates from a combination of original data sources and other documented analyses; this approach results in selecting geometric mean values that are representative of values presented in the source documents. For example, as shown in BSC Table 6-2 (2004ap), DOE evaluated the data for soil-to-plant transfer of technetium in leafy vegetables. This data included eight point values ranging from 9.5 to 180. The DOE-derived truncated lognormal distribution ranged from 3.8 to 550 with a geometric mean of 46. Therefore, the DOE approach resulted in a derived distribution that includes the range of values presented in the source documents.

In addition to reviewing DOE's approach, the NRC staff evaluated a subset of transfer factor values for technetium, iodine, neptunium, americium, and plutonium, as provided in BSC Section 6.2.1.2 (2004ap). This subset includes radionuclides the NRC staff identified as risk significant to DOE's TSPA results (TER Section 2.2.1.3.14.3). The NRC staff compared DOE's values with values independently selected from the available literature and reported in prior NRC-sponsored documents and analyses (NRC, 1992ae; LaPlante and Poor, 1997aa). The

NRC staff noted that DOE-derived geometric mean values were within reasonable ranges of the NRC reported values. For example, considering the prior example of DOE's transfer factor for technetium in leafy vegetables where the source data ranged from 9.5 to 180, the values from NRC (1992ae) and LaPlante and Poor (1997aa) were 44 and 76, respectively. These results compare favorably with the geometric mean of 46 that DOE chose. On the basis of similar evaluations conducted for the remaining radionuclides in the evaluated subset, the NRC staff notes the DOE-derived geometric mean values are within reasonable ranges of the NRC reported values. The NRC staff notes that DOE has provided a reasonable technical basis for the soil-to-plant transfer factors used in the biosphere model. This is based on the results of NRC staff's review of (i) DOE's approach to deriving geometric mean soil-to-plant transfer factors and (ii) the resulting geometric mean values for a subset of the factors.

DOE's approach to propagating uncertainty in the assumed lognormal distributions of soil-to-plant transfer factors generated ranges that, as represented in the technetium-derived lognormal distribution example, encompass the values reported in the source documents. The NRC staff therefore notes that DOE's TSPA biosphere calculations have a reasonable technical basis. By deriving a distribution that encompasses the values reported in the source documents, DOE ensured that parameter sampling in the ERMYN code selects input parameter values from a distribution that encompasses the range of values which are reported in the source documents. Because the source documents are representative of the available scientific data on soil-to-plant transfer factors, this approach results in biosphere dose conversion factor calculations that use soil-to-plant transfer factors based on available scientific data.

For direct deposition of radionuclides on plant surfaces, the plant uptake model in the ERMYN code calculates the radionuclide concentrations in crops from leaf uptake and retention of intercepted irrigation water and dust (SAR Section 2.3.10.3.1.3). These calculations are based on the irrigation deposition rate or dust deposition rate, the fraction of irrigation that originated from above-plant spraying, the crop interception fraction, the translocation factor (fraction of deposited radionuclides that are absorbed and move to other parts of the plant), the weathering half-life (removal rate of contaminants from leaves), the crop growing time, and the crop yield, as identified in SAR Section 2.3.10.3.1.3 and SNL Sections 6.4.3.2 and 6.4.3.3 (2007ac).

The NRC staff's review of DOE's sensitivity and uncertainty analysis of the groundwater biosphere dose conversion factors, as provided in SNL Section 6.13 and Table 6.13-3 (2007ac), showed that these direct deposition inputs had relatively low, or no, effect on biosphere dose conversion factor distributions. Therefore, the NRC staff conducted a general review of these inputs to verify that DOE provided reasonable technical bases. This review noted that DOE has provided reasonable technical bases for the direct deposition input parameters in the ERMYN code. The NRC staff noted that DOE's sensitivity analysis methods involved statistical correlation analyses of individual input parameter distributions with radionuclide-specific biosphere dose conversion factor distributions. These methods are reasonable because they are statistical analysis methods that are commonly used to quantify the relationship between individual model input parameter variability and the variability in model output.

Thus, on the basis of the information discussed in this subsection, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values, and has provided reasonable technical bases for parameter ranges, probability distributions, and bounding values used in the plant uptake submodel in its performance assessment.

Animal Uptake Submodel

SAR Section 2.3.10.3.1.4 described DOE's ERMYN code animal uptake submodel. This submodel calculates radionuclide concentrations in human food products that are derived from livestock that ingest contaminated food and water. For the purpose of modeling, DOE identified four distinct animal product groups: meat, milk, eggs, and poultry. The animal product radionuclide concentrations were calculated on the basis of estimated animal intakes of radionuclides from contaminated feed, water, and soil, as applicable to the groundwater or volcanic eruption modeling cases. Animal feed radionuclide concentrations are computed by the plant uptake submodel (e.g., DOE assumes cows eat locally grown forage and chickens eat local grain). As discussed in BSC (2004ap), DOE used animal product transfer coefficients as input parameters for the fraction of an animal's daily intake of a radionuclide that is transferred to a unit mass or volume of produced food product. DOE's animal product transfer coefficients were selected using the same methods (BSC, 2004ap) described in the previous subsection for soil-to-plant transfer factors.

The NRC staff evaluated DOE's technical bases and supporting data for the selected values and uncertainty ranges for the animal product transfer coefficients used in ERMYN code to calculate biosphere dose conversion factor input parameters for the TSPA model. Data sources DOE used to derive the animal product transfer coefficient input parameters included (i) commonly referenced original data compilations that evaluated a variety of peer-reviewed studies and other technical reports and (ii) other technical analyses that reported soil-to-plant transfer factors selected or derived from available source data or compilations. These references provide a technical basis for selecting generally applicable animal product transfer coefficient values because they include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa) as well as a variety of technical reports authored by various radiological assessment practitioners, including the NRC. Therefore, the referenced documents are representative of the available scientific data on soil-to-plant transfer factors. The documents were reviewed and compiled by experts in the field and provide a reasonable body of technical information to support DOE's derivation of input parameters for the biosphere model.

On the basis of DOE's derivation of animal product transfer coefficients for the biosphere model, DOE's approach of averaging applicable point estimates from a combination of original data sources and other documented analyses is a reasonable one, as it results in selecting geometric mean values that are representative of values presented in the source documents. For example, the data DOE evaluated [provided in BSC Table 6-39 (2004ap)] for the transfer of technetium from feed to meat include 14 point values ranging from 1.0×10^{-4} to 8.7×10^{-3}. DOE derived truncated lognormal distribution ranges from 6.9×10^{-6} to 1.8×10^{-1} with a geometric mean of 1.1×10^{-3}. Therefore, the DOE approach resulted in a derived distribution that included the range of values presented in the source documents.

In addition to the review of DOE's approach, the NRC staff evaluated a subset of DOE's transfer factor values for technetium, iodine, neptunium, americium, and plutonium, as identified in BSC Section 6.3.3 (2004ap). This subset included radionuclides the NRC staff identified as risk significant to DOE's TSPA results (TER Section 2.2.1.3.14.3). The NRC staff compared the DOE geometric mean values with values independently selected from the available literature and reported in prior NRC documents and analyses (NRC, 1992ae; LaPlante and Poor, 1997aa). For example, the values for technetium from NRC (1992ae) and LaPlante and Poor (1997aa) are 8.5×10^{-3} and 1.0×10^{-4}, respectively. The NRC staff notes that these data are reasonably comparable to the geometric mean of 1.1×10^{-3} that DOE derived. When used in

DOE's dose model, DOE's higher value in this comparison is more conservative than the NRC reported values because it would transfer more radionuclides to meat and therefore increase the dose to the reasonably maximally exposed individual relative to the NRC-reported values. On the basis of similar evaluations conducted for the remaining radionuclides in the evaluated subset, the DOE-derived geometric mean values are within reasonable ranges of the NRC-reported values. After reviewing DOE's approach to deriving geometric mean animal product transfer coefficients and the resulting geometric mean values for a subset of the coefficients reviewed, the NRC staff notes that DOE has provided a reasonable technical basis for the animal product transfer coefficients used in the biosphere model.

In reviewing DOE's approach to propagating uncertainty in the assumed lognormal distributions of animal product transfer coefficients, the NRC staff noted that the approach generated ranges that encompass the values reported in the source documents and therefore has a reasonable technical basis for use in TSPA biosphere calculations. DOE assumed a truncated lognormal distribution using the geometric standard deviation computed from the source data and applied a 99 percent confidence interval approach similar to that used for deriving parameter distributions for soil-to-plant transfer factors. By deriving a distribution that encompassed the values reported in the source documents, DOE ensured that parameter sampling in the ERMYN code selects input parameter values from a distribution that encompasses the range of values that are reported in the source documents. Because the source documents are representative of the available scientific data on animal product transfer coefficients, this approach resulted in biosphere dose conversion factor calculations that use animal product transfer coefficients based on available scientific data.

In summary, on the basis of the information discussed in this subsection, DOE has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, and bounding values used in the animal uptake submodel in its performance assessment.

Fish Uptake Submodel

The ERMYN code fish uptake submodel (SAR Section 2.3.10.3.1.5) calculates radionuclide concentrations in fish raised in local fish farms that are assumed to use contaminated groundwater. The input parameter that most influences the results of this model is the bioaccumulation factor. This element-specific factor relates the concentration of radionuclides in the edible portion of the fish to the concentration of radionuclides in the contaminated water in which the fish is submerged. DOE selected bioaccumulation factors on the basis of a review of the applicable literature. DOE's review included fish in all portions of the food chain as well as bottom-feeding fish. DOE assumed a lognormal distribution. For the fish farms noted in Amargosa Valley during DOE's consumption survey, the fish were fed commercial feed that is not locally derived. Feed that is not locally derived would therefore not be expected to become contaminated with radionuclides from a Yucca Mountain release scenario. Therefore, DOE applied a bioaccumulation factor (that accounts for fish ingesting contaminated food and water) to a Yucca Mountain biosphere calculation. Actual conditions suggest that only the water would be contaminated and the analysis will overestimate the fish intake and thereby overestimate the radionuclide concentration in fish. The resulting dose to the reasonably maximally exposed individual from fish consumption is therefore expected to be overestimated.

The NRC staff evaluated the applicability of DOE's technical bases and supporting data for the selected point values and uncertainty ranges for the bioaccumulation factors used in the fish uptake submodel. Data sources DOE used to derive input parameters included

(i) a commonly referenced original data compilation that evaluated a variety of technical reports and some peer-reviewed studies and (ii) other technical analyses that reported bioaccumulation factors selected or derived from available source data or compilations. The NRC staff notes that the referenced documents are representative of available scientific data on fish bioaccumulation factors. The references were reviewed and compiled by experts in the field and include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa) as well as a variety of technical reports authored by various radiological assessment practitioners including the NRC. Therefore, these references provide a technical basis for the selection of generally applicable fish bioaccumulation values.

In reviewing bioaccumulation factor values, the NRC staff notes that DOE's approach of averaging applicable point estimates from a combination of original data sources and other documented analyses resulted in selection of geometric mean values that are representative of values presented in the source documents. For example, the values for fish uptake of carbon DOE evaluated included eight point values ranging from 4.6×10^3 to 5.0×10^4 L/kg [5.5×10^2 to 6×10^3 gal/lb], with a DOE-derived geometric mean of 1.6×10^4 L/kg [1.9×10^3 gal/lb], as identified in BSC Table 6-64 (2004ap). On the basis of the importance of carbon and the fish pathway in DOE's TSPA results relative to other radionuclides and pathways (TER Section 2.2.1.3.14.3), the NRC staff evaluated DOE's supporting information for the fish bioaccumulation factor for carbon. The NRC staff compared the derived value with values independently selected from the available literature (NRC, 1992ae; International Atomic Energy Agency, 1994aa). The NRC staff noted that the DOE-derived geometric mean value was within a reasonable range of the NRC- and the International Atomic Energy Agency-reported values. The values from NRC (1992ae) and the International Atomic Energy Agency (1994aa) are 4.6×10^3 and 5.0×10^4 L/kg [5.5×10^2 to 6×10^3 gal/lb], respectively, and compare favorably with the geometric mean of 1.6×10^4 L/kg [1.9×10^3 gal/lb] DOE derived. Because the data sources DOE used included studies of fish in natural ecosystems that are contaminated with radionuclides, bioaccumulation factors evaluated in those documents would include contributions to fish uptake from contaminated food. Consideration of nonlocally derived feed, which would be uncontaminated, would reduce bioaccumulation factors. The NRC staff notes that DOE's use of these factors leads to an overestimation of fish uptake, thus generating a conservative result.

The NRC staff reviewed DOE's approach to propagating uncertainty in the assumed lognormal distributions of fish bioaccumulation factors and notes that the approach generated ranges that encompass the values reported in the source documents. This is considered reasonable for use in TSPA biosphere calculations. DOE assumed a truncated lognormal distribution using the geometric standard deviation computed from the source data and applying a 99 percent confidence interval approach. As an example, the DOE-derived lognormal distribution of fish bioaccumulation factors for technetium ranged from 3.3 to 120 L/kg [0.4 to 14 gal/lb], which encompasses the range of point values {15 to 78 L/kg [1.8 to 9.4 gal/lb]} in DOE's source documents.

The NRC staff's review of the fish bioaccumulation factors identified an apparent transcription error in the DOE report (BSC, 2004ap). The geometric mean of the fish bioaccumulation factor for carbon was reported differently in two separate tables that should have contained the same values. Specifically, BSC Table 6-64 (2004ap), which is the table that initially derives the value from source data, showed a geometric mean fish bioaccumulation factor for carbon of 1.6×10^4 L/kg [1.9×10^3 gal/lb]. BSC Table 6-65 (2004ap) listed a different value of 4.6×10^3 L/kg [5.5×10^2 gal/lb] for this same parameter (a factor of 3.5 lower than the original value computed in Table 6-64). DOE used the lower value reported in Table 6-65 in the ERMYN calculations, as indicated by SAR Table 2.3.10-10. To evaluate the significance of this discrepancy, the NRC staff considered whether using the lower value would significantly affect DOE's dose results. This evaluation considered that DOE's fish consumption dose scales linearly with the bioaccumulation factor. The NRC staff also evaluated the TSPA results for the 10,000-year simulation period in SAR Figure 2.4-20, which shows that the carbon dose contributes approximately 20 percent of the peak mean total dose. SAR Table 2.3.10-11 indicated that the fish pathway contributed 59 percent to the carbon dose.

The NRC staff's evaluation of these results of DOE's TSPA shows that correcting the geometric mean bioaccumulation factor would not significantly change the all-radionuclide total TSPA results.

In summary, on the basis of the information discussed in this subsection, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided reasonable technical bases for parameter ranges, probability distributions, and bounding values used in the fish uptake submodel in its performance assessment.

Air Submodel

The ERMYN air submodel (SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2) models radionuclide concentrations in air from (i) resuspension of contaminated soil or ash, (ii) evaporative cooler aerosols from the use of contaminated groundwater, and (iii) radon gas emanation from contaminated soil or ash. Inhalation of resuspended particulates is the predominant exposure pathway for Pu-239, Pu-240, and Pu-242 in DOE's performance assessment. Resuspended particulate exposure is modeled in both the groundwater and volcanic ash exposure scenarios. Particulate inhalation also contributes approximately 21 percent to the groundwater dose from Np-237 in the DOE model (SAR Table 2.3.10-11). The other air pathways in DOE's model contribute less to the performance assessment results than particulate inhalation, but are represented in the radionuclides that contribute most to DOE's results as summarized in TER Tables 16-1 and 16-2. Therefore, the NRC staff's detailed review of the technical bases for input parameters and ranges in the air submodel discussed in this TER section evaluated those parameters in DOE's biosphere transport calculations involving particulates, evaporative cooler aerosols, and radon gas.

An important input parameter in the DOE calculation of air concentration of particulates is the mass loading factor {g/m^3 [lb/ft^3]} based on the DOE biosphere dose conversion factor sensitivity analysis results documented in SNL, p. 6-150 (2007ac) and SAR Table 2.3.10-17. The NRC staff reviewed the DOE sensitivity analysis methods and notes that the DOE approach, which includes a statistical correlation analysis of sampled ERMYN model input and biosphere dose conversion factor distribution output, reasonably quantifies the correlation between model input variation and output variation as a means of identifying the sampled input parameters that most influence model results. These methods are reasonable because they are statistical analysis methods that are commonly used to quantify the relationship between individual model input parameter variability and the variability in model output. The mass loading factor computes the concentration of radionuclides in air {Bq/m^3 [Ci/m^3]} from the estimated concentration of radionuclides deposited on the soil surface {Bq/g [Ci/g]}.

In SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2 and supporting references, DOE described its derivation of separate mass loading factors for each exposure scenario (i.e., irrigated soil, volcanic ash) on the basis of available literature. DOE derived individual mass loading input parameters for reasonably maximally exposed individual activity level (active and inactive) and environment (outdoor, indoor, or away from potentially contaminated areas) in its inhalation exposure model.

The NRC staff evaluated DOE's technical bases and supporting data for the mass loading input parameters used in biosphere exposure scenarios involving groundwater and volcanic ash. DOE detailed these in BSC (2006ad).

In reviewing DOE's technical bases for groundwater exposure scenario mass loading input parameter values, the NRC staff evaluated the supporting data DOE used to derive mass loading input parameters against independent NRC estimates derived from available technical information. On the basis of its review of the supporting data, the NRC staff notes that DOE evaluated a reasonable range of studies in published, peer-reviewed literature that measured airborne concentrations of total suspended particulate and PM_{10} {small suspended particles that are less than 10 micrometers [3.9×10^{-4} in] in diameter} for a variety of environments and surface-disturbing activities. The NRC staff notes that DOE based its soil mass loading values on site-specific studies that included measurements of airborne dust in Amargosa Valley applicable to various surface-disturbing activities, including walking, pitching hay, driving, working near construction equipment, and dog walking. DOE also considered a variety of other studies from sites that the NRC staff notes are representative or analogous to Yucca Mountain regional conditions. These include arid or semiarid environments and rural agricultural dust-generating activities and exposure conditions. The NRC staff notes that DOE's supporting information for the mass loading values provides a broad base of technical support that addresses the effects of a range of dust-generating activities and site-specific conditions.

As part of the NRC staff's review of DOE's groundwater exposure scenario mass loading values, the NRC staff evaluated the magnitude of DOE's selected values. This evaluation involved comparing the results of two of the NRC staff's calculations of the mass of soil the reasonably maximally exposed individual inhaled (LaPlante, 2010aa). In these calculations, the mass of soil inhaled is the product of constant values for input parameters for mass loading, exposure time, and breathing rate. One of these calculations was based on DOE's mass loading values, and the other calculation was based on the input parameters the NRC staff derived for the Total-System Performance Assessment 5.1 code by evaluating the available peer-reviewed and other scientific literature (Leslie, et al., 2007aa). Exposure time and breathing rate input parameters in both calculations were set to the same values (Leslie, et al., 2007aa) to isolate the effect of the differences in mass loading inputs on the mass of soil inhaled. Due to the large number of individual mass loading input parameters DOE used, this calculation efficiently evaluated the combined effect of DOE's mass loading parameter choices on an intermediate result in the inhalation dose calculation (i.e., mass of soil inhaled). This comparison showed the calculated daily mass of soil resuspended and inhaled based on the DOE mass loading values was 2.5 times larger than the same result computed using the NRC staff's derived mass loading values. This indicates DOE's selected mass loading values are more conservative than the values that the NRC staff independently derived from available scientific data. For example, the magnitude of DOE's derived values for mass loading, when evaluated in the context of their effect on dose to the reasonably maximally exposed individual (i.e., using the calculation of soil mass inhaled), produce dust inhalation results that are greater than results based on independently derived mass loading and other applicable input parameters, as identified in Leslie, et al., Table 17-1 (2007aa). Therefore, DOE's methods and technical bases for these input parameters are reasonable.

For the volcanic ash scenario, the NRC staff recognizes that limited data are applicable to mass loading for a volcanic eruption in the Yucca Mountain region or for analogous conditions elsewhere. DOE reviewed literature that included measured dust levels of volcanic ash resuspended in air for ambient and surface-disturbing conditions at various sites where volcanoes had recently erupted (within 5 years) and also compared the relevance of each analog site (including the Soufrière Hills Volcano in Montserrat, British West Indies, and the Mt. Spurr Volcano in Alaska) to expected conditions in the Yucca Mountain region.

The NRC staff's review of DOE's technical bases for volcanic ash exposure scenario mass loading input parameter values evaluated the applicability of the supporting data DOE used and the methodology used to derive mass loading input parameters. The NRC staff also evaluated the magnitude of DOE's values against independent NRC estimates derived from available technical information. The NRC staff notes that DOE's consideration of a range of studies that included dust-level measurements taken during a variety of surface-disturbing conditions at volcanic eruption sites provides reasonable technical support for its derived mass loading values.

The NRC staff's review of the magnitude of DOE's volcanic ash exposure scenario mass loading values involved comparing two NRC staff calculations of the mass of resuspended ash the reasonably maximally exposed individual inhaled (LaPlante, 2010aa). In these calculations, the mass of ash inhaled is the product of constant values for input parameters for mass loading, exposure time, and breathing rate. One of these calculations was based on DOE's mass loading values, and the other calculation was based on the input parameters NRC staff derived for the Total-System Performance Assessment 5.1 code by evaluating the available peer- reviewed and other scientific literature, as documented in NRC Table 17-1 (Leslie, et al., 2007aa). Exposure time and breathing rate input parameters in both calculations were set to the same values (Leslie, et al., 2007aa) to isolate the effect of the differences in mass loading inputs on the mass of ash inhaled. Due to the large number of individual mass loading input parameters DOE used, this calculation efficiently evaluated the combined effect of DOE's mass loading parameter choices on an intermediate result in the inhalation dose calculation (i.e., mass of ash inhaled). This comparison showed the calculated daily mass of resuspended, inhaled ash based on the DOE mass loading values was consistent with the same result computed using the mass loading values the NRC staff independently derived from available scientific data. Therefore, the magnitude of DOE's derived values for mass loading, when evaluated in the context of their effect on dose to the reasonably maximally exposed individual (i.e., using the calculation of the mass of ash inhaled), produces dust inhalation results that are within a reasonable range of results based on independently derived mass loading values and other applicable input parameters, as documented in NRC Table 17-1 (Leslie, et al., 2007aa). This independent verification of DOE's derived mass loading input parameters further supports DOE's methods and technical bases for these input parameters.

The NRC staff reviewed DOE's treatment of uncertainty and variability in mass loading values for both exposure scenarios and notes that DOE has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges and probability distributions used in the air submodel. The NRC staff notes that DOE's approach for developing parameter distributions is reasonable, because its values are supported by applicable scientific studies. Although supporting data are limited, DOE reviewed enough information to derive input parameter ranges and a mode[1] to characterize simple distributions for use in the TSPA code.

[1]A mode is a statistic of central tendency for a set of values that represents the value that occurs most frequently in that set.

DOE accomplished this by considering the range of values from the available literature and the applicability of each study to the Yucca Mountain exposure scenarios in terms of similar surface-disturbing activities, an arid or semiarid environment, and measurements of total suspended particulates. DOE then addressed uncertainty and variability in the mass loading parameters [provided in BSC Sections 6.2 and 6.3 (2006ad)] by deriving triangular parameter distributions. These distributions were based on its assessment of the range of applicable literature values and the central tendencies in the data that support selection of a value for the mode of each distribution. DOE conducted similar literature-based evaluation and selection of mass loading ranges and modes to characterize triangular input distributions for each activity environment and for groundwater and ash exposure scenarios. The resulting input distributions were provided in BSC Table 7-1 (2006ad).

DOE evaluated personal exposure measurements of total suspended particulates collected during farming activities at 10 farms near Davis and Sacramento, California, that supported a range of 0.30 to 7.93 mg/m^3 [8.1 × 10^{-6} to 2.1 × 10^{-4} oz/yd^3]. After evaluating additional data from 7 other studies involving mostly farming activities and 22 sets of measurements taken in Amargosa Valley for various types of activities, a range of 1 to 10 mg/m^3 [2.7 × 10^{-5} to 2.73.6 × 10^{-4} oz/yd^3] {with a mode of 3 mg/m^3 [8.1 × 10^{-5} oz/yd^3]} was derived for the ERMYN input for the TSPA analyses for mass loading in the active outdoor environment for groundwater-based biosphere dose calculations. The mode of 3 mg/m^3 [8.1 × 10^{-5} oz/yd^3] was the mean of the maximum mass loading values measured for 22 surface-disturbing activities in Amargosa Valley, as identified in BSC Section 6.2.1.3 (2006ad).

The NRC staff considers these mass loading data reasonable for inclusion in the biosphere model of Yucca Mountain for three reasons: (i) the data describe activities that are consistent with the characteristics of the Yucca Mountain region (e.g., farming, arid conditions), (ii) the data are based on a subset of the measurements are taken from the Yucca Mountain region, and (iii) the data include breathing zone sampling measurements. The breathing zone sampling data are particularly relevant for supporting an inhalation exposure scenario because the measurements were taken in air within the breathing zone. On the basis of the NRC staff's review of DOE's methods for deriving uncertainty distributions in the mass loading values, the NRC staff notes that DOE has addressed uncertainty and variability in the mass loading parameter values, and has thus provided reasonable technical bases for parameter ranges and probability distributions used in the air submodel in the ERMYN code.

As discussed in SAR Section 2.3.10.3.1.2, the air submodel in the ERMYN code also calculates indoor air radionuclide concentrations from aerosols released from evaporative coolers. This calculation is based on the concentration of radionuclides in groundwater, the rate of water evaporation from coolers, the indoor air exchange rate, and the fraction of radionuclides in water that transfer to air (the water-to-air transfer fraction). DOE identified the water-to-air transfer fraction in the evaporative cooler model as an important input parameter in the evaporative cooler calculation. The NRC staff evaluated the derivation of the water-to-air transfer fraction and notes that values of the water-to-air fraction were selected conservatively to bound possible values. DOE assumed a uniform concentration ratio distribution from 0 to 1 for dissolved solids and 1 for gases on the basis of a lack of available studies on contaminant aerosols from evaporative coolers. The NRC staff notes that this value is conservative because dissolved solids do not evaporate when water evaporates (the same process is used to purify water by distillation). The assumed distribution causes the model to release an average of 50 percent of the dissolved solids (including dissolved radionuclides) that are in the groundwater directly to indoor air. This increases the inhalation dose from aerosols beyond what would be expected under actual conditions. On the basis of the information reviewed, the NRC staff notes that

DOE has addressed uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, and bounding values used in the aerosol release from evaporative cooler calculation within the ERMYN air submodel.

Airborne concentrations of radon gas released from soils irrigated with contaminated water or from contaminated volcanic ash were also calculated in the air submodel (SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2; SNL, 2007ac; BSC, 2004ap). Both exposure scenarios consider indoor and outdoor radon concentrations. In the volcanic ash scenario, the outdoor air concentration is also used for the indoor air concentration because DOE expected the outdoor concentration to be higher than the indoor concentration as a result of the small radon contribution from ash below the reasonably maximally exposed individual's house. DOE's groundwater scenario calculates separate radon concentrations for indoor and outdoor environments. DOE's indoor radon concentration calculations evaluated radon released from soil beneath a hypothetical house built on land that was previously irrigated by contaminated water. In this model, the rate of radon released into the house is a proportion of the outdoor radon flux that accounts for diffusion of radon from underlying soil through the foundation. Indoor radon concentrations in the model were calculated based on (i) the radon flux into the house from soil beneath the house and from outdoors, (ii) the interior air exchange rate, and (iii) the interior volume of the house. The interior air exchange rates account for periods of evaporative cooler use and nonuse based on increased ventilation during cooler operation, which decreases radon concentration in air. The indoor radon diffusion methods are consistent with those used in the RESRAD dose assessment code (Yu, et al., 2001aa) that the U.S. Environmental Protection Agency (EPA) developed. Outdoor radon concentrations are based on factors that relate the airborne concentration of Rn-222 to either (i) the Ra-226 concentration in the soil for the groundwater scenario or (ii) the Rn-222 flux density for the volcanic ash scenario.

The NRC staff evaluated DOE's technical bases and supporting data for input parameters used in the indoor and outdoor radon concentration modeling in the ERMYN code. Input parameters that were reviewed included the fraction of radon flux entering the foundation from soil, and home ventilation rates. DOE chose the concentration fraction of the radon flux from soil underneath the house that would diffuse into the house to be uniformly distributed from 0.1 to 0.25 on the basis of measurements in homes with concrete foundations (SAR Section 2.3.10.3.1.2). The home ventilation rates (for evaporative cooler nonuse periods) were based on minimum ventilation recommendations for manufactured homes, data from a survey of approximately 3,000 U.S. homes, and information from a trade organization representing home ventilation equipment manufacturers (BSC, 2004ap). The home ventilation rates for evaporative cooler use were estimated on the basis of cooler flow rates and the average home interior volume. Uncertainty and variability in the ventilation rates were propagated by deriving a truncated lognormal distribution on the basis of the survey data (for no cooler use) and a uniform distribution for cooler use ventilation rates that spans the estimated range (BSC, 2004ap). Regarding DOE's radon concentration calculation, the NRC staff notes that DOE used diverse information sources (including information applicable to manufactured homes, national survey data of home ventilation rates, and ventilation equipment information) and further notes that DOE has provided a reasonable technical basis for deriving input parameters for that calculation.

The NRC staff notes that the approaches used to derive these distributions use common methods that result in distributions that reasonably represent the values reported in the referenced information. On the basis of the review of the supporting documentation, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges and probability distributions used in the air submodel radon concentration calculations. The radon concentrations are used for calculating biosphere dose conversion factor input parameters for the TSPA code.

On the basis of information discussed in this subsection, including the NRC staff's review of the data supporting selected input parameters and distributions that DOE used to model the inhalation of resuspended soil and ash, aerosols from evaporative coolers, and radon released from irrigated soils, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided reasonable technical bases for parameter ranges, probability distributions, or bounding values used in the air submodel in its performance assessment.

2.2.1.3.14.3.3 Assessment of Human Exposure

DOE calculated human exposures from estimated concentrations of radionuclides in groundwater and soil in three exposure submodels of the ERMYN code (SNL, 2007ac). These submodels are the primary exposure pathways addressed in the DOE exposure scenarios and include the external exposure submodel, the inhalation exposure submodel, and the ingestion exposure submodel (SAR Sections 2.3.10.3.1.7, 2.3.10.3.1.8, and 2.3.10.3.1.9 for the groundwater exposure scenario and SAR Sections 2.3.10.3.2.5, 2.3.10.3.2.6, and 2.3.10.3.2.7 for the volcanic ash exposure scenario). Considering the biosphere pathways that are the primary contributors to dose to the reasonably maximally exposed individual (TER Tables 16-1 and 16-2), the NRC staff's risk-informed review focused on the inhalation and ingestion exposure submodels. These exposure submodels compute the reasonably maximally exposed individual's annual intake of radionuclides {e.g., Bq/yr [Ci/yr]} on the basis of the environmental media concentrations (e.g., air, water, livestock products, fish) calculated by the biosphere environmental transport submodels that are depicted in SAR Figures 2.3.10-9 and 2.3.10-10, described in SAR Section 2.3.10.3, and evaluated in TER Sections 2.2.1.3.14.3.2 and 2.2.1.3.12.3. DOE's exposure models also convert the calculated reasonably maximally exposed individual radionuclide intakes to dose. The NRC staff's review of DOE's conversion of intakes into dose is evaluated in TER Section 2.2.1.3.14.3.4.

Inhalation Exposure Model

DOE's ERMYN inhalation exposure submodel calculates reasonably maximally exposed individual radionuclide intakes by modeling the inhalation of contaminated air. In the model, airborne contaminants include resuspended soil or ash particulates, aerosols from evaporative coolers, or radon gas emanating from contaminants in soil or ash. DOE calculations showed that the inhalation exposure pathways that are the most risk significant contributors to DOE's performance assessment results (TER Tables 16-1 and 16-2) are resuspended particulates from soil and aerosols from evaporative coolers. Inhalation of gaseous emissions of radon from contaminants in soil also contributes to DOE's long-term dose calculation results, but that contribution is less than the contributions from particulates and aerosols.

DOE's exposure calculations for these three inhalation exposure pathways involve exposure time and breathing rate input parameters that are the focus of the NRC staff's review. The NRC staff focused on these parameters because (i) they directly influence the calculated dose; (ii) they are influenced by a number of other complex variables including the types of human activities, activity durations, and intensity of physical activity; (iii) they are used in all three of the inhalation exposure pathways; and (iv) DOE's documented technical bases for these inputs are particularly complex relative to the other inhalation exposure submodel inputs such as the fraction of houses with coolers, the evaporative cooler use factor, and the equilibrium factor for radon decay products.

DOE's exposure time input parameters in the ERMYN inhalation exposure submodel apportion the amount of time the reasonably maximally exposed individual spends in various environments where exposure could occur into three categories: outdoor, indoor, and away from areas potentially contaminated by Yucca Mountain activities. The DOE inhalation exposure submodel also apportions the amount of time spent conducting surface-disturbing activities, which DOE identified as active, to account for increased exposure to radionuclides resuspended from the ground surface by the activity. DOE identified the amount of time that the reasonably maximally exposed individual was not conducting surface-disturbing activities as inactive. DOE grouped the Amargosa Valley population into four population categories: nonworkers, commuters, local outdoor workers, and local indoor workers. DOE apportioned time spent into five activity–environment categories (by combining the three environment categories with the two activity-level categories): active outdoors, inactive outdoors, active indoors, inactive indoors (sleep), and away from areas potentially contaminated by Yucca Mountain activities. Exposure times were derived from census information on age distribution; employment; commuting characteristics of Amargosa Valley, Nevada, residents; and national survey data on activity times (BSC, 2005ab).

The NRC staff evaluated DOE's technical bases and supporting data for the exposure times that were used to derive activity–environment categories (BSC, 2005ab). DOE provided a derivation of exposure times on the basis of data from surveys of the Amargosa Valley and national populations. Year 2000 census data from the Amargosa Valley census county division (Bureau of the Census, 2002aa) provided the population distribution by age, work status and hours worked, commute time, and industry of employment. DOE also used detailed national survey data on activity time budgets from Klepeis, et al. (1996aa) and EPA (1997aa) to assign the fraction of time spent inside a residence; outdoors; in a vehicle; and at stores, restaurants, and other indoor locations.

The NRC staff notes that DOE's data reasonably support DOE's derivation of exposure times. This is based on DOE's use of local and national data obtained from sources including the Bureau of the Census and the EPA to derive exposure times using a data synthesis approach. DOE's data synthesis approach used percentages of time spent conducting activities at various locations by age group with Amargosa Valley population information to generate percentages applicable to the Amargosa Valley population. DOE then used additional national survey results (EPA, 1997aa) on time spent outdoors to derive active and inactive outdoor exposure times and apportioned the resulting times spent at various locations to derive its exposure time input parameter values. DOE's estimates for the fraction of outdoor activity that includes surface-disturbing activities (20 percent of public outdoor time and 50 percent of construction worker outdoor time) were based on EPA survey data and information on local practices. The NRC staff evaluated a sample of values used in the derivation against source documents to verify that the values were incorporated accurately. The NRC staff also conducted simplified calculations of the documented results to verify the data synthesis and found no discrepancies or errors. DOE lognormal distributions of exposure time estimates to propagate variation in the ERMYN code were based on the standard errors provided with the survey data and the application of standard lognormal distribution statistical methods. For some parameters, DOE intentionally assigned standard errors that were larger than those associated with the national survey data to account for uncertainties in applying national data to local conditions. These uncertainties result from potential differences in local human activity practices compared to national patterns and the effect of differences in survey sample sizes on standard errors.

Because propagating variation from survey data directly quantifies empirical data variability, the NRC staff notes that DOE has accounted for uncertainty and variability in exposure time parameter values and has provided reasonable technical bases for parameter values, ranges, and probability distributions used in the ERMYN inhalation exposure submodel. The NRC staff also notes that the exposure time input parameters used in the inhalation exposure calculations are consistent with the living styles of current Amargosa Valley residents, because the parameters were accurately derived from survey information applicable to the local population.

DOE derived breathing rates for each population group and for each level of activity within the four potentially contaminated exposure environments (active outdoors, inactive outdoors, active indoors, and inactive indoors). DOE combined breathing rate information for adults by gender and level of physical activity from International Commission on Radiological Protection (1994aa) with census demographic information for Amargosa Valley to derive population gender-weighted breathing rates. DOE then used information from International Commission on Radiological Protection (1994aa) on the fraction of daily time devoted to different levels of activity to derive breathing rates for each exposure environment category. In this manner, the exposure environment categories applied to all population groups considered in the model and were derived, in part, on the basis of surveys of Amargosa Valley, Nevada, residents.

The NRC staff evaluated DOE's technical bases and supporting data for the breathing rates detailed in BSC (2005ab). The gender-weighted breathing rate values DOE calculated from International Commission on Radiological Protection breathing rates and census data for Amargosa Valley were 0.39, 0.47, 1.38, and 2.86 m^3/hr [0.51, 0.61, 1.81, and 3.74 yd^3/hr] for sleep, sitting, light exercise, and heavy exercise, respectively. These values compare favorably with EPA-recommended adult breathing rate values for use in short-term exposure calculations (EPA, 1997aa). The EPA values are 0.4, 0.5, 1.0, 1.6, and 3.2 m^3/hr [0.5, 0.7, 1.3, 2.1, and 4.2 yd^3/hr] for a similar progression of increasing activity level including rest, sedentary, light, moderate, and heavy activities, respectively. The EPA value for heavy activity {3.2 m^3/hr [4.2 yd^3/hr]} is somewhat higher than the DOE value. However, the EPA values apply to short-term exposures that would have higher breathing rates than values used for the long-term exposure calculations (i.e., annual dose). Additionally, the EPA-recommended value for outdoor workers for heavy activity {2.5 m^3/hr [3.3 yd^3/hr]} is lower than the DOE value.

The NRC staff also reviewed DOE's methods for deriving the final set of breathing rates for each of the four exposure environments. These methods involved use of a weighted sum approach to compute a single breathing rate value applicable to all population groups for each exposure environment. The NRC staff notes that DOE's methods [BSC Table 6-15 (2005ab)] are reasonable for use in long-term exposure calculations in the ERMYN biosphere inhalation exposure model. DOE used these breathing rates in ERMYN calculations as individual fixed input parameters for each exposure environment, and the model propagates breathing rate input parameter uncertainty or variability due to differences in activity level by using different exposure environments. Because the DOE approach accounts for variability and uncertainty in breathing rates based on human activity level and the magnitude of the breathing rates compare favorably with other data sources NRC staff identified as discussed previously, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided the technical bases for the breathing rates used in the performance assessment.

In summary, on the basis of the information discussed in this subsection, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the inhalation exposure submodel in its performance assessment.

Ingestion Exposure Model

DOE's ingestion exposure submodel in ERMYN calculates radionuclide intakes by modeling reasonably maximally exposed individual consumption of contaminated water, crops, animal products (milk, meat, poultry, eggs, fish), and soil. Ingestion pathways have a more pronounced effect on DOE's performance assessment results during the initial 10,000-year period than on the results for the 1-million-year period because of the radionuclides that dominate the dose calculations. Exposure pathways that contribute most to DOE's performance assessment results (TER Section 2.2.1.3.14.3) include drinking water, fish consumption, and animal product consumption (milk, meat, eggs). The exposure calculations for these ingestion exposure pathways involve consumption rate input parameters for modeled food products the reasonably maximally exposed individual consumes (i.e., water, fish, and animal food products including milk, meat, and eggs).

DOE derived food consumption rates used in the ERMYN ingestion exposure submodel from a DOE-sponsored survey of Amargosa Valley residents (DOE, 1997ab). The exception is DOE's modeling of drinking water consumption, which DOE assumed to be 2 L/day [0.53 gal]. The Amargosa Valley survey measured how often residents consumed various locally produced food products. The calculated arithmetic mean annual consumption rates for various food types and corresponding standard deviations were then used as input parameters for sampling parameter values in ERMYN assuming a lognormal distribution, as detailed in BSC Section 6.4.2, Table 6-21 (2005ab).

The NRC staff evaluated DOE's technical bases and supporting data for the food and water consumption rates as detailed in BSC (2005ab). The supporting data included a survey of 195 of the reported 872 adult Amargosa residents (22 percent of the population) to obtain information on local food consumption frequency. The survey yielded useful responses from 187 of these contacts and provided explanations for the 8 responses that were eliminated from further analysis. The NRC staff's review of the survey notes that DOE reasonably generated data that provide a technical basis for deriving consumption rate input parameters for the biosphere calculations.

The NRC staff reviewed DOE's methods for calculating the annual consumption rates for locally produced food on the basis of the survey data, census information (to incorporate more recent Amargosa Valley population information), and national average daily intakes from the U.S. Department of Agriculture. The resulting values for annual food consumption rates appear to be low compared to values based on the more commonly used national food consumption surveys (e.g., NRC, 1992ae). This is because DOE's values are population averages of individual consumption rates that apply specifically to the consumption of locally produced foods [detailed in BSC Figures 6-3 through 6-12 (2005ab)] rather than all food consumed. Additionally, a large number of Amargosa Valley residents who do not consume locally produced food weight the average consumption in the population to lower values than are found in national average food consumption rates. The low local food consumption rates also reflect the limited capacity of agricultural food production in the Amargosa Valley (LaPlante and Poor, 1997aa). The NRC staff notes that the methods used by DOE are reasonable for supporting the dose calculations. The NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided technical bases for consumption rate parameter values, ranges, and probability distributions used in the ingestion exposure submodel in its performance assessment.

In summary, on the basis of the information discussed in this subsection, the NRC staff notes that DOE has accounted for uncertainty and variability in parameter values and has provided reasonable technical bases for parameter ranges, probability distributions, or bounding values used in the ingestion exposure submodel in its performance assessment.

2.2.1.3.14.3.4 Assessment of Dosimetry

The DOE biosphere model uses dose coefficients from the Federal Guidance Report 13 (EPA, 1999aa), which uses tissue-weighting factors recommended in International Commission on Radiological Protection, Publication 60 (1991aa) to calculate effective dose from both internal and external radiation sources. In its TSPA modeling, DOE identified 28 primary radionuclides that were the primary contributors to dose to the reasonably maximally exposed individual using a radionuclide screening analysis (SNL, 2007ac; SAR Section 2.3.7.4.1.2). DOE then converted radionuclide intake or external exposure to dose using the dose coefficients for the 28 primary radionuclides. DOE used dose coefficients for external exposure that are defined as the effective dose rate per unit radionuclide concentration in the soil. DOE also used dose coefficients for inhalation and ingestion as the committed effective dose per unit radionuclide intake by inhalation or ingestion.

DOE chose dose coefficients for intake of radionuclides used in the biosphere model for adults and a total effective dose equivalent commitment period of 50 years. The biokinetic and dosimetric models used to develop these dose coefficients are based on a hypothetical average adult person with the anatomical and physiological characteristics the International Commission on Radiological Protection (1975aa) defined with further modifications as described in Federal Guidance Report 13 (EPA, 1999aa). DOE used breathing rates in its biosphere model that are based on the more recent biometric results for adults from the respiratory tract model the International Commission on Radiological Protection (1994aa) developed, as discussed in TER Section 2.2.1.3.14.3.3. The NRC staff notes that DOE used reasonable scientific models and methodologies (recommended by the International Commission on Radiological Protection) to calculate the total effective dose equivalent.

For ingestion and inhalation, DOE typically chose dose coefficients for the chemical form of the radionuclide that resulted in the highest dose to avoid underestimating dose. In a few cases, such as C-14, DOE chose dose coefficients that were consistent with the form that was being transported [i.e., the gaseous (carbon dioxide) and solid (particulate) forms have different dose coefficients]. The NRC staff notes that this approach is reasonable because using these dose coefficients would not result in underestimating dose to the reasonably maximally exposed individual.

DOE calculated the total effective dose equivalent to the reasonably maximally exposed individual as the sum of the effective dose equivalent from external sources plus the committed dose equivalent from internal sources (i.e., sources either inhaled or ingested). This approach is reasonable, as it is consistent with International Commission on Radiological Protection recommendations for calculating the total effective dose equivalent.

The description that DOE provided in the SAR is adequate to fully assess the dosimetry models used in the TSPA. The NRC staff performed a detailed review of the dosimetry data for a selection of radionuclides, including Tc-99, C-14, I-129, Ra-226, Pu-239, Pu-240, Pu-242, and Np-237. The review also included comparing a sample of dose coefficients for other radionuclides that were used in the TSPA. No discrepancies were found between the dose coefficients included in the biosphere model report and those tabulated in Federal Guidance

Report 13 (EPA, 1999aa). The dose coefficients DOE used are reasonable because they incorporate the most current and appropriate scientific models and methodologies for an adult receptor.

In summary, on the basis of the information discussed in this subsection, the NRC staff notes that DOE has provided reasonable technical bases for values used in the dosimetry model in its performance assessment. DOE's dosimetry method uses the current and appropriate scientific models and methodologies (recommended by the International Commission on Radiological Protection). DOE's dose coefficients were developed on the basis of an adult reasonably maximally exposed individual metabolic and physiologic data that is consistent with present knowledge.

2.2.1.3.14.3.5 Assessment of Integrated Biosphere Modeling Results

DOE biosphere modeling results were provided in SAR Section 2.3.10 and analyzed in greater detail in the Biosphere Model Report (SNL, 2007ac). The exposure pathways determined to be the most risk significant in the DOE performance assessment varied depending on the particular radionuclide. The radionuclides and pathways that were most risk significant in the DOE TSPA calculations are summarized in TER Table 16-1 for the 10,000-year simulation period and TER Table 16-2 for the 1-million-year simulation period.

To validate the integrated biosphere model, DOE compared the calculation results for each environmental transport and exposure submodel of the ERMYN code with comparable calculation results from five other biosphere transport and exposure process-level models (SAR Section 2.3.10.5). DOE stated that the results of the process-level calculations used in those other models were the same, or similar, to the results obtained using the biosphere model (SNL, 2007ac). To verify implementation, DOE compared the results of the biosphere model for representative radionuclides (Pu-239, Ra-226, Th-232, and C-14) with the results of spreadsheet calculations—on the basis of equations used in the biosphere mathematical model—and the results were identical (SNL, 2007ac). On the basis of the information reviewed, DOE's model support calculations provide reasonable technical bases for the biosphere models used in the performance assessment.

The NRC staff also performed confirmatory calculations to further assess the ERMYN model for a subset of the radionuclides that DOE's performance assessment identified as the most risk significant (TER Tables 16-1 and 16-2) (LaPlante, 2010aa). These confirmatory calculations were performed for the drinking water ingestion pathway of the groundwater exposure scenario. Drinking water ingestion was chosen because (i) it was identified as one of the most risk-significant pathways in the DOE performance assessment, (ii) the model is not complex and therefore could be executed efficiently, and (iii) the results of the confirmatory calculations could be directly compared to DOE results. The results of the confirmatory groundwater calculations were within 2 percent of the mean biosphere dose conversion factor results DOE reported in SAR Table 2.3.10-2, weighted by the DOE drinking water ingestion pathway fractions in SAR Table 2.3.10-11. Given that the DOE results are summary statistics of the model output, a difference of only 2 percent confirms that both calculations were consistent. The NRC staff notes that DOE's groundwater calculations provide reasonable technical bases for the biosphere models used in the performance assessment.

In SAR Section 2.3.10.4.1, DOE described how its performance assessment considered and evaluated alternative conceptual models for seven processes in the biosphere model abstraction. These processes are (i) radon release from soil, (ii) transfer of radionuclides to air from evaporative cooler operation, (iii) transfer of radionuclides to plants from direct deposition of irrigation water, (iv) transfer of airborne particulates to plants from direct deposition, (v) transfer of radionuclides to animal products from livestock inhalation, (vi) transfer of C-14 to crops, and (vii) environment-specific inhalation exposure. DOE's consideration of alternative conceptual models provides reasonable technical bases for the biosphere models used in the performance assessment.

2.2.1.3.14.4 NRC Staff Conclusions

The NRC staff notes that DOE's description of this model abstraction for the biosphere characteristics is consistent with the guidance in the YMRP. NRC staff also notes that the technical approach for the biosphere characteristics discussed in this chapter is reasonable for use in the Total System Performance Assessment (TSPA).

2.2.1.3.14.5 References

BSC. 2006ad. "Inhalation Exposure Input Parameters for the Biosphere Model." ANL–MGR–MD–000001. Rev. 04. ACN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2005ab. "Characteristics of the Receptor for the Biosphere Model." ANL–MGR–MD–000005. Rev. 04. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

BSC. 2004ap. "Environmental Transport Input Parameters for the Biosphere Model." ANL–MGR–MD–000007. Rev. 02. ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Bureau of the Census. 2002aa. "2000 Summary File 3 (SF 3) Sample Data, Amargosa Valley CCD, Nye County, Nevada." ANL–MGR–MD–000005, Rev. 4, DIRS 159728. Washington, DC: U.S. Department of Commerce, Bureau of the Census.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 1997ab. "The 1997 Biosphere Food Consumption Survey Summary Findings and Technical Documentation." ANL–MGR–MD–000005. Rev. 4. DIRS 100332. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

EPA. 1999aa. "Federal Guidance Report 13, Cancer Risk Coefficients for Environmental Exposure to Radionuclides." EPA 402–R–99–001. Washington, DC: U.S. Environmental Protection Agency.

EPA. 1997aa. "Exposure Factors Handbook, Volume III, Activity Factors." EPA/600/P–95/002Fc. Washington, DC: U.S. Environmental Protection Agency.

International Atomic Energy Agency. 1994aa. "Handbook of Parameter Values for the Prediction of Coefficients: Workers and Members Radionuclide Transfer in Temperate Environments." Technical Report Series No. 364. Vienna, Austria: International Atomic Energy Agency.

International Commission on Radiological Protection. 1994aa. “Human Respiratory Tract Model for Radiological Protection.” ANL–MGR–MD–000005. Rev. 4. DIRS 153705. *Annals of the ICRP.* Publication 66. Vol. 24, No. 1-3. New York City, New York: Pergamon Press.

International Commission on Radiological Protection. 1991aa. “1990 Recommendations of the International Radiological of Commission Protection.” ANL–MGR–MD–000005. Rev. 4. DIRS 153705. *Annals of the ICRP*. Publication 60. Vol. 21, No. 1-3. New York City, New York: Pergamon Press.

International Commission of Radiological Protection. 1975aa. “Report of the Task Group on Reference Man.” *Annals of the ICRP*. Publication 23. Vol. 23. New York City, New York: Pergamon Press.

Klepeis, N.E., A.M. Tsang, and J.V. Behar. 1996aa. “Analysis of the National Human Activity Patter Survey (NHAPS) Respondents From a Standpoint of Exposure Assessment, Percentage of Time Spent, Duration, and Frequency of Occurrence for Selected Microenvironments by Gender, Age, Time-of-Day, Day-of-Week, Season, and U.S. Census Region—Final Report.” EPA/600/R–96/074. ANL–MGR–MD–000005. Rev. 4. DIRS 159299. Washington, DC: U.S. Environmental Protection Agency, Office of Research and Development.

LaPlante, P. 2010aa. "Calculations Supporting the Biosphere Characteristics Section, Volume 3, Technical Evaluation Report for the NRC Review of the DOE Yucca Mountain Repository License Application." Electronic Scientific Notebook 1056E. San Antonio, Texas: CNWRA.

LaPlante, P.A. and K. Poor. 1997aa. “Information and Analyses to Support Selection of Critical Groups and Reference Biospheres for Yucca Mountain Exposure Scenarios.” CNWRA 97-009. San Antonio, Texas: CNWRA.

Leslie, B., C. Grossman, and J. Durham. 2007aa. “Total-system Performance Assessment (TPA) Version 5.1 Module Descriptions and User Guide.” Rev. 1. ML072710060. San Antonio, Texas: CNWRA.

NRC. 2009ab. “Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63.” Published March 13, 2009. ML090850014. Washington, DC: NRC.

NRC. 2005aa. NUREG–1762, “Integrated Issue Resolution Status Report.” Rev. 1. Washington, DC: NRC.

NRC. 1992ae. NUREG–5512, “Residual Radioactive Contamination From Decommissioning: Technical Basis for Translating Contamination Levels To Annual Total Effective Dose Equivalent—Final Report.” Vol. 1. Washington, DC: NRC.

SNL. 2008ab. “Features, Events, and Processes for the Total System Performance Assessment: Analyses.” ANL–WIS–MD–000027. Rev. 00. ACN 01, ERD 01, ERD 02. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2008ag. “Total System Performance Assessment Model/Analysis for the License Application.” MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

SNL. 2007ac. "Biosphere Model Report." MDL–MGR–MD–000001. Rev. 02. ERD 01. Las Vegas, Nevada: Sandia National Laboratories.

Yu, C., A.J. Zielen, J.-J. Cheng, D.J. LePoire, E. Gnanapragasam, S. Kamboj, J. Arnish, A. Wallo, W.A. Williams, and H. Peterson. 2001aa. "User's Manual for RESRAD Version 6." ANL/EAD–4. Argonne, Illinois: Argonne National Laboratory.

(Intentionally Left Blank)

CHAPTER 17

2.2.1.4.1 Postclosure Individual Protection Calculation

2.2.1.4.1.1 Introduction

This chapter of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) Total System Performance Assessment (TSPA) for the individual protection calculation, as presented in DOE's Safety Analysis Report (SAR) Section 2.4.2 (DOE, 2008ab). DOE conducted an analysis, through its TSPA computer model, that evaluates the behavior of the high-level waste repository in terms of an annual dose due to potential releases from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a repository at Yucca Mountain. In particular, the NRC staff's review evaluates whether (i) the performance assessment analysis includes the appropriate scenario classes [a set or combination of features, events, and processes (FEPs) that the performance assessment model uses to represent a class or type of scenario such seismic activity], (ii) the representation of the scenario classes within the performance assessment is credible (e.g., the performance assessment results are consistent with the models, parameters, and assumptions that make up the performance assessment), and (iii) the annual dose the performance assessment estimates is statistically stable.

2.2.1.4.1.2 Evaluation Criteria

The regulations specify that a performance assessment must be used to evaluate the individual protection calculation. According to 10 CFR 63.2, performance assessment is defined as an analysis that

- Identifies the FEPs (except human intrusion) and sequences of events and processes (except human intrusion) that might affect the Yucca Mountain disposal system and their probabilities of occurring during 10,000 years after disposal

- Examines the effects of those FEPs and sequences of events and processes upon the performance of the Yucca Mountain disposal system

- Estimates the dose incurred by the reasonably maximally exposed individual (RMEI), including the associated uncertainties, as a result of releases caused by all significant FEPs, and sequences of events and processes, weighted by their probability of occurrence

The requirements for developing performance assessment analyses (e.g., consideration of features, events, and processes included in the performance assessment; determination of event probabilities; and consideration of uncertainties) are relevant to previous TER Sections 2.2.1.2 and 2.2.1.3. These previous sections evaluate DOE's development of the analytic models used in the performance assessment analysis. The requirements also specify how the performance assessment model is used to estimate the annual dose to the reasonably maximally exposed individual (RMEI). In general, the performance assessment requirements are to

- Estimate the annual dose as a result of releases caused by all significant features, events, and processes weighted by their probability of occurrence (10 CFR 63.2)

- Calculate the arithmetic mean (i.e., average) of the annual dose for the period within 1 million years after disposal (10 CFR 63.303)

The NRC staff's review followed the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). The acceptance criteria address the following:

- Scenarios used in the calculation of the annual dose as a function of time are adequate.

- Total System Performance Assessment provides a credible representation of repository performance.

- The annual dose to the reasonably maximally exposed individual (RMEI) is adequately demonstrated.

2.2.1.4.1.3 Technical Evaluation

2.2.1.4.1.3.1 Introduction

DOE's performance assessment is implemented through its TSPA code. The TSPA code is used to represent the range of behavior of a Yucca Mountain repository, accounting for uncertainty in the features, events, and processes (FEPs) that could affect the repository evolution. DOE developed its analysis of repository performance using distinct groupings of FEPs—referred to as "scenario" or "event" classes. In very general terms, there are two broad categories of scenario classes: nominal and disruptive. The nominal scenario class comprises those FEPs that are present under "normal" conditions (e.g., infiltration of water, corrosion of the waste package, release of radionuclides, transport of radionuclides in groundwater). The disruptive scenario class includes additional FEPs that account for the effects of specific events (i.e., seismic events, volcanic activity, fault movement) that disrupt or alter the repository performance differently from what the nominal scenario class portrays. In DOE's TSPA model, the nominal scenario class is considered part of the seismic ground motion modeling case so that the combined effects of waste package corrosion, which degrades the mechanical strength of the waste package, and mechanical damage of the waste package due to seismic ground motion are appropriately considered in the post-10,000-year period (SAR p. 2.4-36). (Note: For the initial 10,000 years, the nominal scenario class does not result in any dose, as detailed in SAR p. 2.4-62 and Figure 2.4-22a.) A key aspect of the disruptive scenario classes is the consideration of the probability or likelihood that the disruptive event will occur. The annual dose is weighted by the probability of its occurrence.

The NRC staff reviewed SAR Sections 2.4.1 and 2.4.2 and the Total System Performance Assessment (TSPA) model files including intermediate results provided as part of the SAR. Additionally, the NRC staff's review in this chapter relies on the NRC staff's reviews, presented in the previous 16 chapters, on the multiple barriers, scenarios, event probabilities, and model abstractions implemented in DOE's TSPA model (TER Sections 2.2.1.1–2.2.1.3.14). Specific TER chapters, as applicable, are referenced in this chapter.

The NRC staff's review entails

- Determining that the probabilities and consequences of each of the scenario classes are appropriately included in the average annual dose (TER Section 2.2.1.4.1.3.2)

- Determining that the results of the performance assessment provide a credible representation of repository performance [e.g., the intermediate results, such as waste package failure, and release rates from the engineered barrier system (EBS), unsaturated, and saturated zones, are consistent with the model abstractions and the average annual dose; confirmatory calculations are consistent with the performance assessment results] (TER Section 2.2.1.4.1.3.3)

- Determining that the calculated average annual dose is statistically stable [e.g., increasing the number of simulations (statistical sample size) performed with DOE's TSPA is not expected to significantly change the average annual dose] (TER Section 2.2.1.4.1.3.4)

2.2.1.4.1.3.2 Scenarios Used in Calculation of Annual Dose

2.2.1.4.1.3.2.1 Summary of DOE Approach

DOE has identified three distinct event scenario classes (sometimes referred as event classes) that are included in its TSPA model for the individual protection calculation: (i) early failures, (ii) seismic events, and (iii) igneous events. DOE has used two modeling cases within each scenario class to represent specific aspects of the scenario. The early failure scenario class is composed of an early waste package failure modeling case and an early drip shield failure modeling case. The seismic scenario class is composed of a seismic ground motion modeling case and a seismic fault displacement modeling case. The igneous scenario class is composed of an igneous intrusion modeling case and a volcanic eruption modeling case.

DOE's average annual dose curve for individual protection (SAR Figure 2.4-18) is determined by summing the effects of all the scenario classes (i.e., early failure, seismic, and igneous). The annual doses attributed to each of the scenario classes are a direct result of the features, events, and processes used to represent the scenario class and its probability of occurrence.

Features, events, and processes included in the scenario classes are reviewed in the NRC staff model abstraction review (TER Sections 2.2.1.3.1–2.2.1.3.14). The NRC staff also reviewed and evaluated the features, events, and processes that DOE considered and excluded from the performance assessment (see TER Section 2.2.1.2.1). As previously stated in the TER, DOE's approach for identifying the appropriate FEPs used to represent the scenario classes is reasonable. The NRC staff's reviews are presented in the previously identified TER Sections 2.2.1.2.1 and 2.2.1.3.1–2.2.1.3.14.

Scenario Class Probabilities

DOE's TSPA assessment incorporates the following three distinct event scenario classes: (i) the igneous activity scenario class, which has a very low annual probability [on the order of a 1 in 100 million chance of occurring per year, as outlined in CRWMS M&O (1996aa)]; (ii) the seismic scenario class, which typically results in numerous events occurring over 1 million years (according to SAR Section 2.4.2.1.6, p. 2.4-50, seismic events are expected to occur frequently; however, it is important to evaluate the timing and magnitude of seismic events);

and (iii) the early failure scenario class, for which there is a low probability of occurrence for an individual waste package (SAR Section 2.4.2.1.6, p. 2.4-49). These three event scenario classes include the occurrence of nominal processes, whereas the nominal scenario class represents repository behavior in which no events occur (i.e., no seismic events, no igneous events, and no early failure events; see SAR Section 2.4.2.1.3, pp. 2.4-30–31). DOE has described how its approach to combine the scenarios to derive aggregated annual dose estimates is appropriate in that it tends to slightly overestimate dose by double counting waste packages potentially affected by different failure modes from the different scenarios (e.g., waste packages failed by a seismic event and an igneous event would be double counted; see SAR Section 2.4.2.1.7).

Igneous Scenario Class

Probability

The igneous scenario class is composed of an igneous intrusion modeling case and a volcanic eruption modeling case. The probability for the igneous intrusion modeling case is described in the DOE model as a Poisson process (a random process in which the events occur independently of one another), and intrusive events are distributed in time with a mean recurrence frequency of 1.7×10^{-8} per year, with a 5th and 95th percentile uncertainty spanning nearly two orders of magnitude, 7.4×10^{-10} to 5.5×10^{-8} per year. DOE describes the probability of the volcanic eruption modeling case as a subset of the probability used for the igneous intrusion modeling case by using a conditional probability that an igneous intrusive event will also have an eruptive component that ejects waste into the atmosphere. The conditional probability is composed of (i) a conditional probability of 0.28 that an igneous intrusive event could have an eruptive component and (ii) a conditional probability of 0.2968 that the eruptive component of an igneous intrusive event could intersect the waste packages. The combination of these two probabilities results in a net conditional probability of 0.083 that an igneous intrusive event would also manifest a volcanic eruption that intersects waste packages in the repository footprint. Thus, the mean recurrence frequency for the volcanic eruption modeling case is 1.4×10^{-9} per year based on the mean recurrence frequency provided previously for the igneous intrusion modeling case (i.e., 1.7×10^{-8} per year).

Igneous Event Contribution to Average Annual Dose Curve

DOE evaluated the igneous intrusion modeling case and the volcanic eruption modeling case separately assuming the entire intact or degraded repository inventory is available for release for each modeling case. A decrease in the repository inventory due to the occurrence of other scenarios is not considered. DOE stated this is a conservative assumption in that it does not underestimate the annual dose to the reasonably maximally exposed individual (RMEI) [SAR Section 2.4.2.1.7.4, p. 2.4-53].

The average annual dose from the igneous scenario class is the sum of the contributions from the igneous intrusion modeling case and the volcanic eruption modeling case. The average annual dose from the igneous intrusion modeling case is more than 99 percent of the average annual dose from the igneous scenario class (intrusion plus volcanic eruption) (SAR Figure 2.4-18).

The igneous intrusion modeling case is the second largest contributor to the overall average annual dose (i.e., summation of average annual dose from all scenario classes) in the 10,000-year period and the largest contributor to the overall average annual dose after

10,000 years (SAR Figure 2.4-18). In the 10,000-year period, Tc-99, Pu-239, Pu-240, and I-129 radioactive elements or radionuclides are the dominant contributors to the average annual dose. After 10,000 years, the dominant radionuclide contributors to the average annual dose are Pu-239, Pu-242, Np-237, and Ra-226 (SAR Figure 2.4-30).

Early Failure Scenario Class

Probability

The early failure scenario class includes two modeling cases: early drip shield failure and early waste package failure. Failure of the drip shield allows seepage water to contact a waste package. Failure of the waste package allows release of radionuclides out of the waste package by diffusion and/or advection. Early failure of either the drip shield or waste package is associated with undetected defects accounting for manufacturing processes such as improper heat treatment, base metal selection flaws, improper weld filler material, and emplacement errors. DOE assumed that all of the waste packages under early failed drip shields would also be considered failed if contacted by seepage water, as described in SNL Section 6.4.1, p. 6.4-4 (2008ag). The probability of having a large number of drip shields and waste packages fail early due to undetected defects is very small. On average, the number of waste packages affected by early failure of drip shields and waste packages is less than 0.02 percent of the total number of waste packages (SAR Section 2.4.2.1.7.2, p. 2.4-52).

Early Failure Contribution to Average Annual Dose Curve

The early failure scenario class contributes on the order of 1 percent or less to the overall average annual dose curve (SAR Figure 2.4-18). In particular, the average annual dose for the early drip shield failure modeling case, which includes contributions from waste packages under drip conditions and unprotected by the drip shields against contact with seepage water, is below 1×10^{-5} mSv [0.001 mrem] for all times. The average annual dose for the early waste package failure modeling case is generally 10 times or more greater than the average annual dose for the early drip shield failure modeling case but still on the order of 100 times less than the overall average annual dose curve (SAR Figure 2.4-18).

Seismic Scenario Class

Probability

The seismic scenario class is composed of a seismic ground motion modeling case and a seismic fault displacement modeling case. The DOE model describes seismic ground motion events as a Poisson process, with events distributed in time with a maximum mean recurrence frequency of 4.287×10^{4} per year {corresponding to the frequency of events with a peak ground velocity exceeding 0.219 m/s [0.72 ft/s] (SAR Section 2.4.2.1.6, p. 2.4-50)}. This is the maximum recurrence frequency of seismic ground motion events that could result in repository damage. Given a recurrence frequency of 4.287×10^{-4} per year, it is expected that seismic ground motion events of that magnitude could occur, on average, every 2,200 years. Thus, because it is expected that multiple seismic events will occur during the 1-million-year period, DOE considered cumulative effects of seismic ground motion from multiple events.

The DOE model describes seismic fault displacement events as a Poisson process with events distributed in time, with a maximum mean recurrence frequency of 2×10^{-7} per year. Because multiple seismic fault displacement events that could affect the repository are expected to be sufficiently rare and therefore inconsequential to repository performance, DOE evaluated only the effects of single seismic fault displacements.

In both modeling cases for the seismic scenario class, the consequence of events that have a recurrence frequency between the maximum mean recurrence frequency (i.e., those which could cause repository damage) down to the probability cutoff of 1×10^{-8} per year are evaluated. The magnitude of an individual seismic event is determined through the use of a probabilistic seismic hazard analysis (PSHA) curve. The PSHA was developed primarily through the use of an expert elicitation process and is documented in CRWMS M&O (1998aa) (see also TER Sections 2.2.1.2.2.3.2 and 2.2.1.3.2 for further details on the NRC staff evaluation).

Seismic Event Contribution to Average Annual Dose Curve

The average annual dose from the seismic scenario class is the sum of the contributions from the two seismic modeling cases: (i) the seismic ground motion modeling case, which addresses the potential for seismic events to damage waste packages and drip shields due to vibratory ground motion and (ii) the seismic fault displacement modeling case, which addresses the effects of fault displacement on waste packages and drip shields.

Nominal corrosion processes have the potential to alter the susceptibility of the waste package to damage during seismic ground motion events as the corrosion processes gradually weaken the mechanical strength of the waste package. Therefore, the seismic ground motion modeling case also includes both waste package degradation from the nominal processes (e.g., general corrosion) and seismic ground motion.

The average annual dose from the seismic ground motion modeling case is at least 10 times larger than the average annual dose from the seismic fault displacement modeling case over the entire 1-million-year period (see SAR Figure 2.4-18). Although the average annual dose curve from the seismic ground motion modeling case also includes the effects from the nominal scenario class, the nominal scenario class contributes no more than 50 percent to the seismic ground motion modeling case average annual dose curve (compare SAR Figures 2.4-18 and 2.4-22).

The seismic ground motion modeling case is second only to the igneous intrusion modeling case in overall significance to the overall average annual dose curve. The seismic ground motion modeling case is the largest contributor to the overall average annual dose curve for the period after 1,500 years through 20,000 years. The overall average annual dose curve in either the initial 10,000 years or after 10,000 years is dominated by contributions from the seismic ground motion modeling case and the igneous intrusion modeling case.

2.2.1.4.1.3.2.2 NRC Staff Evaluation of Scenarios Used in Calculation of Annual Dose

The NRC staff has reviewed DOE's analytic models and assumptions used in its TSPA analyses, as documented in the SAR and supporting documents, and notes that DOE's inclusion of the annual dose from each of the scenario classes into the overall average annual dose curve is reasonable for the following reasons.

- The two modeling cases (i.e., igneous intrusion modeling case and seismic ground motion modeling case) that result in the greatest number of failed waste packages are the largest contributors to the overall average annual dose curve (see TER Section 2.2.1.4.1.3.3.1.1.2.

- DOE's overall average annual dose curve appropriately includes the probabilities of occurrence for the scenario classes (the values for probability of igneous and seismic activity are appropriate and reflect the uncertainty for the occurrence of these events; see TER Sections 2.2.1.2.2.3.1 and 2.2.1.2.2.3.2).

- The value for the probability of early failures reflects the uncertainty in potential defects regarding the drip shield and waste package (see TER Section 2.2.1.2.2.4).

- Model assumptions of the early failure scenario class tend to overestimate release consequences in the sense that no credit is given to early failed waste packages to impede contact of water or moisture with waste forms. Additionally, no credit is assigned to an early failed drip shield as a barrier against seepage, and the corresponding exposed waste packages are also given no credit to contain or impede contact of waste forms with seepage if located under seepage conditions (see TER Section 2.2.1.3.1.3.1.2 for the drip shield and TER Section 2.2.1.3.1.3.2.4 for the waste package).

- The consequences for an eruptive igneous event are appropriate and reflect conservative assumptions in certain key areas of uncertainty such as the amount of waste entrained in the tephra (volcanic ash) (see TER Section 2.2.1.3.13).

- The consequences for an intrusive igneous event are appropriate and reflect conservative assumptions in certain key areas of uncertainty such as all waste packages and drip shields being rendered ineffective (see TER Section 2.2.1.3.10).

- The magnitude of seismic events leading to waste package damage is appropriate and reflects the material properties of the engineered barriers and design (see TER Section 2.2.1.3.2).

- The number of waste packages that can be potentially affected by fault displacement is appropriate and reflects the geologic setting of Yucca Mountain and the layout of the repository footprint (see TER Section 2.2.1.3.2.4).

2.2.1.4.1.3.3 Credible Representation of Repository Performance

This section of the TER documents the NRC staff review to determine the credibility of the representation of repository performance in DOE's Total System Performance Assessment (TSPA). In particular, the NRC staff evaluates the consistency of the characteristics of repository performance in the TSPA (e.g., number of waste packages failed, transport of radionuclides in the geosphere, and scenario probabilities) with the overall dose estimated by the TSPA. The focus of the NRC staff review is on those aspects of repository performance that have the most significance to risk (i.e., the probability-weighted dose estimate). The NRC staff review of individual components of DOE's TSPA (i.e., model abstractions; features, events, and processes included in the TSPA; scenario probabilities; and barrier capabilities) is documented in the previous 16 chapters of this Postclosure Volume of the TER.

The NRC staff reviewed the TSPA documentation in SAR Volume 2 and in the TSPA GoldSim computer model and associated computer files (including intermediate results saved in the GoldSim output files). The NRC staff's review of the TSPA analyses considered how the collection of features, events, and processes that are included in the TSPA model represents a credible characterization of the repository. The NRC staff's review approach entails a quantitative evaluation of the attributes of DOE's TSPA calculation that most significantly impact estimating the annual dose to the reasonably maximally exposed individual (RMEI). Identification of the important attributes for performance are based on the NRC staff review of the capabilities of the barriers important to waste isolation (TER Section 2.2.1.1), the model abstractions in the TSPA code (TER Sections 2.2.1.3.1–2.2.1.3.14), and the NRC staff independent analysis with its performance assessment model, as outlined in CNWRA and NRC (2008aa) and NRC Appendix D (2005aa).

DOE has used two modeling cases for each of the three scenario classes [for igneous (intrusive and eruption), for seismic (ground motion and fault displacement), and for early failure (drip shield and waste package)] to estimate overall performance of the Yucca Mountain repository (see Table 17-1). Only one modeling case (volcanic eruptive) releases radionuclides directly to the atmosphere via volcanic ash. The other five modeling cases (seismic ground motion, seismic fault displacement, igneous intrusion, early waste package failure, and early drip shield failure) release radionuclides through groundwater movement. The NRC staff review of DOE's TSPA calculation related to groundwater releases is provided in Section 2.2.1.4.1.3.3.1 and the NRC staff review of DOE's TSPA calculation for the volcanic eruption modeling case is provided in Section 2.2.1.4.1.3.3.2.

2.2.1.4.1.3.3.1 DOE's TSPA Calculation Related to Groundwater Releases

2.2.1.4.1.3.3.1.1 Summary of DOE Approach in TSPA

The modeling cases associated with groundwater releases are described by tracking the water through the system. For example, water could infiltrate the top of the mountain and move downward to the repository; after waste packages are breached and radionuclide releases occur, water could transport radionuclides through the unsaturated zone then through the saturated zone to the location of the reasonably maximally exposed individual (RMEI). In general, the description of the groundwater releases is based on the following repository performance characteristics:

- Seepage of water entering the drifts (tunnels containing the waste packages)
- Damage to engineered barriers (drip shield and waste package)
- Seepage of water into the waste packages
- Release of radionuclides from the waste package
- Transport of radionuclides in the unsaturated zone
- Transport of radionuclides in the saturated zone
- Annual dose to the reasonably maximally exposed individual (RMEI)

The volcanic eruption modeling case evaluates the release of radionuclides via volcanic ash deposited on the ground. The volcanic eruption modeling case is evaluated separately (see "Description and Understanding of TSPA Calculation Related to Releases from a Volcanic Eruption Event" later in this chapter) from the modeling cases that involve radionuclide release through the groundwater pathway.

Table 17-1. Scenario Classes and Modeling Cases Included in DOE's TSPA		
Scenario Class	**Modeling Case**	**Transport Pathway***
Early failure	Drip shield	Groundwater
	Waste package	Groundwater
Seismic	Ground motion	Groundwater
	Fault displacement	Groundwater
Igneous	Intrusive	Groundwater
	Eruption	Atmospheric (volcanic ash)
*Transport pathway indicates the primary pathway for radionuclides to be transported away from the repository to the accessible environment.		

2.2.1.4.1.3.3.1.1.1 Summary of DOE's TSPA for Seepage of Water Into Drifts

The flux of water reaching the drifts (i.e., drift seepage) is originally derived from rainfall over the mountain. Two important metrics for performance for the upper natural barrier are seepage flux (the amount of liquid water entering the repository drifts) and seepage fraction (number of waste package locations with dripping water). The latter is the fraction of the repository area where seepage occurs (the seeping environment); the remainder of the area would not receive seepage. Seepage, or dripping water, has the potential to fall onto the drip shields and later contact the waste packages after the drip shields degrade sufficiently to allow water to pass through the drip shield.

Precipitation to Deep Percolation

DOE divided the first 10,000 years into three periods: present day (0–600 years), monsoonal (600–2,000 years), and glacial transition (2,000–10,000 years) climates (SAR Tables 2.3-1–2.3-4; DOE, 2008ab). The glacial transition climate spans 80 percent of the first 10,000 years and has the most significant impact on the performance of the repository over this initial period. For the glacial period, DOE calculated an average precipitation of 296.7 mm/yr [11.7 in/yr] in the repository footprint, as described in DOE Enclosure 1, Table 1 (2010ai). Processes including runoff of water from hillsides, evaporation, and lateral diversion of water caused by the Paintbrush Tuff nonwelded rock layer alter the amount of rainfall that eventually ends up as deep percolation (the amount of water reaching the repository level). DOE estimated an average deep percolation of 21.74 mm/yr [0.86 in/yr] at the repository horizon, as detailed in DOE Enclosure 1, Table 1 (2010ai) at the repository footprint for the initial 10,000 years. Thus, approximately 7 percent of the rainfall ends up as deep percolation at the repository footprint. (TER Sections 2.2.1.3.5 and 2.2.1.3.6 provide further details on climate and infiltration.)

For the post-10,000-year period, the specified time-independent flux for deep percolation, representing average climate conditions over the long term is a range of 10 to 100 mm/yr [0.39 to 3.9 in/yr] using a truncated lognormal distribution, which results in an arithmetic mean of 37 mm/yr [1.5 in/yr] for the deep percolation. In the SAR, DOE used a log-uniform distribution with an arithmetic mean of 32 mm/yr [1.3 in/yr] and a range between 13 and 64 mm/yr [0.51 and 2.5 in/yr] (SAR Section 2.3.2.3.5.1) on the basis of the proposed regulation. The specified deep percolation range represents a 16 percent higher average deep percolation than the value DOE used in the SAR (see TER Section 2.2.1.3.6 for further discussion on the reason for the different values used for deep percolation). Using this higher deep percolation for a bounding calculation, DOE estimated a corresponding increase in the 1-million-year reasonably maximally exposed individual (RMEI) average annual dose from 0.02 mSv/yr to 0.023 mSv/yr [2.0 mrem/yr to 2.3 mrem/yr], as detailed in DOE Enclosure 6 (2009cb).

Deep Percolation to Seepage

The TSPA model includes a number of factors such as focusing or diverging of flow at the repository footprint, vapor barrier surrounding a drift when the drift temperature is above the boiling point of water, different waste package types, capillary diversion, drift degradation (prevalent in the seismic cases) in estimating seepage fraction, and drift seepage. Seepage is set to zero if the drift wall temperature exceeds 100 °C [212 °F] (SAR Section 2.3.3.4.1.1). The period of time when drift wall temperatures exceed 100 °C [212 °F] is generally limited to the first 2,000 years, as the heat generated by radionuclide decay decreases.

The longevity of the drip shield and waste package, however, limits the significance of the seepage at early times, especially during the initial 10,000 years. The uncertainties of the effects on the longevity of the drip shield and waste package are evaluated in TER Sections 2.2.1.3.1 and 2.2.1.3.2.

Over the repository footprint, seepage flux is approximately 10 percent of the deep percolation for intact drifts and can increase up to 49 percent for degraded drifts as DOE predicted through the seismic ground motion modeling case, outlined in DOE Enclosure 5 (2010ai). For the igneous intrusive scenario, all of the percolating flux enters the drift at all locations, as described in DOE Enclosure 7 (2009ct). Table 17-2 provides the DOE values for the seepage fraction and the average seepage rate over the repository footprint (i.e., averaged over both seeping and nonseeping environments). As the seepage rate increases, the number of locations where dripping occurs (seepage fraction) also increases.

2.2.1.4.1.3.3.1.1.2 Summary of DOE's TSPA for Damage to Engineered Barriers (Drip Shield and Waste Package)

The drip shield and the waste package are two important components of the engineered barrier system (EBS) (TER Sections 2.2.1.3.1 and 2.2.1.3.2 evaluate behavior of the drip shield and waste package). The drip shield degrades gradually over time (from general corrosion) or at specific times from large seismic events or igneous events. From the distributions considered in the TSPA model to represent corrosion rates of Titanium Grade 7, time to failure by general corrosion of the drip shield was computed to range from 260,000 to 340,000 years (SAR Figures 2.1-8 and 2.4-24). Seismic ground motion can collapse the drip shield by mechanical failure due to static and dynamic loads caused by rockfall and ground motion. As illustrated in SAR Figure 2.1-11, drip shield collapse due to seismic ground motion occurs primarily between 25,000 and 350,000 years with the vast majority of the failures occurring between 200,000 and 300,000 years (see SAR Figure 2.4.24). Drip shields are assumed to fail whenever an intrusive igneous event occurs, when a fault displacement event breaches the waste package, or when significant general corrosion occurs. Once the drip shield is failed, seepage water that enters the drifts can contact the surface of the waste package.

Table 17-2. DOE's Mean Values for the Seepage Rate Into Drifts*			
Time Period	**Nominal/Early Failure**	**Seismic Ground Motion**	**Igneous Intrusive**
Seepage from 2,000 to 10,000 years	2.0 (mm/yr)	2.3 (mm/yr)	21.7 (mm/yr)
Seepage fraction from 2,000 to 10,000 years	31%	31%	100%
Seepage after 10,000 years	3.4 (mm/yr)	15.5 (mm/yr)	31.7 (mm/yr)
Seepage fraction after 10,000 years	40%	69%	100%
*See TER Section 2.2.1.3.6 for further information.			

The damage mechanisms leading to waste package failure are "crack" and "patch" (holes) failure. Within DOE's TSPA model, crack and patch failures of the waste package are treated separately because of differences in how water may enter a breached waste package. DOE assumed that seepage water cannot freely flow through cracks on the waste package because of the small size of the cracks. Because of processes such as general corrosion, or ruptures and punctures of the waste package, patch failures represent significantly larger openings and seepage water is assumed to enter the waste package through these holes or openings. The waste packages with large or patch openings could allow release of radionuclides carried by flowing water (i.e., advective release) and diffusion, while those with crack openings could only allow release by diffusion.

The five modeling cases associated with groundwater releases have very distinct characteristics for the timing and extent of waste package failure. Assumed to occur at the time of closure, the early failure scenario class consists of two modeling cases: drip shield early failure and waste package early failure. The early failure scenario has, in probabilistic terms, on average, less than one waste package and one drip shield failing (see TER Section 2.2.1.3.1.3.2.4 for early waste package failure and TER Section 2.2.1.3.1.3.1.2 for drip shield failure). The igneous intrusion modeling case assumes all waste packages fail at the time of the event and lead to release to the water pathway (see TER Section 2.2.1.3.10 for further details). The seismic scenario class consists of a ground motion modeling case, where the waste packages are damaged by seismic ground motion, and a fault displacement modeling case, where displacement along a fault may damage the waste packages that lie along the fault. The seismic fault displacement modeling case has, on average, tens of waste packages failing (see TER Section 2.2.1.3.2 for further details). The seismic ground motion modeling case contains a range of waste package failures—typically initial damage is primarily due to cracks in the waste package from ground motion, and, later in time (e.g., after 100,000 years), general corrosion, ruptures and punctures, and further cracking damages the waste packages.

Table 17-3 provides the cumulative number of waste package failures accounted for in the seismic ground motion modeling case for selected times (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide some perspective on the time-dependent nature of waste package failure. The values in Table 17-3 for the seismic ground motion modeling case were

Table 17-3. Cumulative Number of CSNF* and CDSP† Waste Packages Breached for the Seismic Ground Motion and Igneous Intrusion Modeling Cases in DOE's TSPA					
Process		**10,000 Years**	**100,000 Years**	**400,000 Years**	**800,000 Years**
Seismic Ground Motion	All failure types (cracks and patches)	CSNF 1.6 CDSP 34.2	CSNF 20.5 CDSP 1,024.8	CSNF 739.2 CDSP 1,366.4	CSNF 3,531.6 CDSP 2,049.6
	Ruptures and punctures (patches)	CSNF 0 CDSP 0.3	CSNF 0 CDSP 0.7	CSNF 24.6 CDSP 13.7	CSNF 82.1 CDSP 34.2
	General corrosion (patches)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 0.3 CDSP 4.8	CSNF 328.5 CDSP 239.1
Igneous Intrusion‡	All failures are patch failures	CSNF 1.4 CDSP 0.6	CSNF 14 CDSP 5.8	CSNF 55.9 CDSP 23.2	CSNF 111.7 CDSP 46.5

*Repository contains 8,213 commercial spent nuclear fuel (CSNF) waste packages.
†Repository contains 3,416 codisposal (CDSP) waste packages.
‡Igneous intrusion values calculated assuming all waste packages failed weighted by the probability of occurrence of an igneous event on or before the given time (i.e., annual probability for the event of 1.7×10^{-8} per year multiplied by the time period).

taken from DOE Enclosure 1, Figures 9, 10, 13, and 14 (DOE 2009bj) and SAR Figures 2.1-12 a and c. These values are weighted by the probability for the seismic events to occur. In the first 10,000 years, a small number of waste packages fail in the seismic ground motion modeling case due to rare but potentially damaging earthquakes. The majority of the failed waste packages are codisposal packages (CDSP). These packages are not equivalent to the more robustly designed transportation, aging, and disposal canisters that contain the commercial spent nuclear fuel (CSNF) waste packages. Most of the initial damage is attributed to cracks that are small enough to prevent seepage water from entering the waste package. At 100,000 years, the number of failed codisposal waste packages due to cracks is 1,025 (30 percent of the 3,416 CDSP waste packages in the repository) and the number of failed commercial spent nuclear fuel waste packages is 20 (less than 1 percent of the 8,213 CSNF waste packages in the repository). At 800,000 years, approximately 40 percent of the commercial spent nuclear fuel and 50 percent of the codisposal packages waste packages fail due to cracks. Some of these cracks are from seismically induced stress corrosion cracks on the waste package surface. Others are due to stress corrosion cracks in the closure welds—considered a general corrosion process. Waste packages start to fail by general corrosion patches at around 500,000 years, and 239 (approximately 7 percent) of the codisposal waste packages and 328 (approximately 4 percent) of the commercial spent nuclear fuel waste packages have at least one failed patch due to general corrosion by 800,000 years, as described in DOE Enclosure 1, Response Number 1, Figures 9 and 10 (2009bj). Waste package failure due to ruptures and punctures is limited: approximately 1 percent of the commercial spent nuclear fuel and codisposal waste packages failed after 800,000 years, as shown in DOE Enclosure 1, Response Number 1, Figures 13 and 14 (2009bj).

The majority of the waste package failures are associated with the seismic ground motion modeling case, which includes the nominal processes such as general corrosion, and the igneous intrusion modeling case. In DOE's TSPA model, the seismic ground motion and igneous intrusion modeling cases contribute most to the overall average annual dose curve and are generally more than a factor of 10 greater than the other modeling cases (SAR Figure 2.4-18). The NRC staff review in TER Sections 2.2.1.3.1 and 2.2.1.3.2 notes that DOE's representation for the timing and extent of waste package failures is reasonable. Therefore, the NRC staff detailed review of DOE's groundwater releases focuses on these two modeling cases, which are the dominant contributors to the overall average annual dose curve (SAR Figure 2.4-18).

2.2.1.4.1.3.3.1.1.3 Summary of DOE's TSPA for Seepage of Water Into Waste Packages

In the TPSA model, two conditions are required for seepage water to enter a waste package: (i) drip shield failure must occur to allow water to contact the waste package outer barrier and (ii) the waste package outer barrier must be breached by patches (it is assumed that seepage or dripping water cannot flow into waste package cracks due to the small opening of cracks; therefore, seepage water enters the waste package only through patch failures). When these required conditions are met, water flow through the waste package is modeled as quasi-steady state, where water flux into the waste package is equal to water flux out.

The waste package surface is divided into a large number of patches with each patch having distinct properties that can affect the corrosion rate of the patch. The extent of waste package degradation (i.e., number of patch failures on the waste package) determines the quantity of water entering the waste package. The waste package outer barrier is considered unable to divert water when a mean value of approximately 62 patches fail (62 patches comprises approximately 4 percent of the total surface area of the waste package), at which point water flow through the waste package equals the incoming seepage rate. When fewer than 62 patches fail, water flux through the waste package is linearly related to the number of patches that fail. Generally, a single patch failure allows 1/62 of the seepage flux to pass through the waste package. Because of uncertainty incorporated into the submodel, this value can range from 0 to 2.4 times the 1/62 value.

The waste package patch failures in the seismic ground motion modeling case are of limited extent (i.e., the waste package surface area has a limited number of failed patches). Approximately 1 percent of the waste package surface area is breached by general corrosion after 1 million years, as described in DOE Enclosure 1, Response Number 1, Figures 11 and 12 (2009bj). Patch failure by ruptures and punctures compromises approximately 0.2 percent of the waste package surface by 1 million years, as outlined in DOE Enclosure 1, Response Number 1, Figures 15 and 16 (2009bj). Thus, the compromised area remains limited for the 1-million-year period after patch failure occurs due to general corrosion, ruptures, or punctures. Additionally, the number of waste package patch failures is limited over the 1-million-year time period: 10 percent of waste package failures are from general corrosion and approximately 3 percent of waste package failures from ruptures and punctures after 1 million years. These failures occur primarily at long times [e.g., after 400,000 years, as shown in DOE Enclosure 1, Response Number 1, Figures 9, 10, 13, and 14 (2009bj)].

For the igneous intrusion modeling case, the drip shield and waste package are assumed to be ineffective barriers against seepage in the TSPA model (i.e., no credit to decrease seepage is given to the drip shield or waste package after the time at which the event occurs). Thus, DOE assumes all drip shields and waste packages are failed after the igneous intrusion event occurs [SAR Volume 2, p. 2.4-42; DOE Enclosure 8, p. 13 (2009ct)].

2.2.1.4.1.3.3.1.1.4 Summary of DOE's TSPA for Release of Radionuclides From the Waste Package

Waste form degradation and subsequent radionuclide release cannot occur prior to waste package breach and/or failure. Assuming waste package failure, radionuclide release may be advective if seepage water enters the waste package; otherwise, the release will be a diffusive release.

Because seepage water does not flow through crack failures, the radionuclide release from cracks in the waste package is controlled by radionuclide diffusion in an assumed continuum of aqueous pathways through the cracks. DOE described patch failure as a more extensive damage mechanism of the waste package surface from general corrosion, or waste package ruptures and punctures driven by seismic events and mechanical interactions with drip shields or other waste packages. Radionuclide release through damaged patches is assumed to be diffusive if seepage does not contact the waste package, which could occur in the DOE model if the drip shield is not breached, or if the waste package is under nondrip conditions. If, on the other hand, the waste package is under drip conditions, the drip shield is failed, and the waste package is breached by patches (by corrosion or processes driven by seismic events) then radionuclides are released from the waste package by flowing water (i.e., advective release) and diffusion. Advective release is effective also in the igneous intrusion case, in which all waste packages fail completely and seepage is assumed to contact all waste packages.

In general, diffusive and advective release of radionuclides from a waste package will be affected by the size of the openings, degradation rate of the waste, solubility limits, sorption onto corrosion products, and the presence of colloids. The significance of these features and processes can vary for specific radionuclides, as described next.

Size of the Openings

The overall surface area of the crack and patch openings directly affects radionuclide diffusion out of the waste package (more surface area results in more release). Additionally, the overall surface area of the patch openings can affect the amount of water entering the waste package and thus the amount of dissolved radionuclides released from the waste package in the advective or flowing water. (Note: Once an average of 4 percent of the waste package surface is failed, due to patches, DOE assumes that the waste package no longer limits the amount of seepage water that enters the waste package. Thus, all seepage water is assumed to enter the waste package when 4 percent or more of the waste package surface area is failed.)

Degradation Rate of Waste

Radionuclides cannot leave the waste package faster than the waste degrades. Generally, the degradation rates used in TSPA for CSNF result in somewhat short times (e.g., hundreds to thousands of years) for the waste form to significantly degrade, as outlined in DOE Enclosure 5, Table 1.1-1 (2009an); therefore, the degradation rate only affects those radionuclides that are not limited by other release constraints inside the waste package (e.g., Tc-99 and I-129 are not solubility limited and are not sorbed or attached onto corrosion products). For CDSP waste packages, the glass waste form can degrade much slower than CSNF (e.g., thousands to millions of years for the glass waste form to significantly degrade versus hundreds to thousands of years for CSNF); however, the defense spent nuclear fuel waste form is assumed to instantly degrade, as described in DOE Enclosure 5, p. 6 (2009an).

Solubility Limit

Some radionuclides have a solubility limit—a function of the properties of the radionuclide and the water chemistry inside the waste package—that controls the amount that can be dissolved in water. Radionuclides such as plutonium (e.g., Pu-242) and neptunium (e.g., Np-237) have the potential for low release rates due to solubility limits (see TER Section 2.2.1.3.4).

Corrosion Products

The TSPA includes a process by which certain radionuclides attach onto corrosion products within the waste package, and thus release from the waste package is delayed. This is especially effective for a radionuclide such as Np-237 that is somewhat soluble and attaches onto corrosion products, as described in DOE Enclosure 5, p. 22 (2009an).

Colloids

Colloids can facilitate release of radionuclides out of the waste package; radionuclides sorbed or attached onto irreversible colloids are not affected by solubility limits and stationary corrosion products. Colloids can also sequester radionuclides by becoming unstable (see TER Section 2.2.1.3.4). DOE stated that the contribution to annual dose from irreversible colloids is small [i.e., contribution from irreversible colloids never exceeds 30 percent, as shown in DOE Enclosure 5, pp. 24–25 (2009an)]. The contribution to annual dose from a radionuclide such as Pu-242 will be mainly from aqueous releases, which are composed of both dissolved radionuclides and reversible colloids (e.g., SAR p. 2.4-93 and SAR Figure 2.4-73).
Release rates of radionuclides from an individual waste package are dependent on the type of radionuclide. High mobility characterizes the soluble, nonsorbing radionuclides (e.g., Tc-99, I-129, Cl-36, Se-79), which may result in nearly complete release of the inventory of the high-mobility radionuclides from the engineered barrier system (EBS) over the 1-million-year period (SAR Figure 2.1-24). Much lower mobility characterizes the relatively insoluble, sorbing nuclides (e.g., Np-237, Pu-242). For the concentration-limited radionuclides (e.g., Pu-242 and Np-237), DOE explained that releases will be significantly lower than the release rates for soluble radionuclides (e.g., Tc-99) and will increase as water flow into the waste packages increases. For example, as corrosion patch area increases in size over time, more water may enter the waste package (SAR p. 2.4-63). At the end of the 1-million-year period, approximately 0.1 percent of the Np-237 inventory has been released from the engineered barrier system (EBS) with the majority of the release occurring over the later portion of the 1-million-year period (SAR Figure 2.1-25).

2.2.1.4.1.3.3.1.1.5 Summary of DOE's TSPA for Transport of Radionuclides in the Unsaturated and Saturated Zones

Transport of radionuclides through the unsaturated zone is affected by the following processes: (i) relatively fast fracture flow versus slow flow in the porous rock matrix, (ii) radionuclides that sorb onto mineral surfaces, and (iii) colloid-facilitated transport of radionuclides.

Transport of radionuclides can depend significantly on whether water flow occurs principally in fractures or in porous media. Flow in fractures is conceptualized as being relatively fast because the effective porosity is relatively small [average estimated value of 0.001

(SAR Table 2.3.9-4)]. Conversely, flow in porous media is conceptualized as relatively slow because the effective flow porosity is relatively high [average estimated value of 0.18 (SAR Table 2.3.9-4)]. Additionally, given the limited surface area for fracture surfaces as compared to rock pores, radionuclides can be significantly delayed by sorption to mineral surfaces.

Colloids are tiny particles that remain suspended in water and are thus able to move with the water and facilitate the transport of certain radionuclides. Transport times of strongly sorbing radionuclides, such as plutonium and americium, can decrease (e.g., increase transport velocity) by permanently attaching onto colloids. This colloid attachment can also occur reversibly when radionuclides temporarily attach to or detach from colloids as they move through the system. Generally, most of the release of Pu-242 from the unsaturated and saturated zone is via dissolved plutonium and plutonium reversibly associated with colloids (referred to as "aqueous" release in SAR Figure 2.4-108). Limited release of Pu-242 is associated with irreversible colloids, whereby Pu-242 permanently attaches onto the colloid.

Transport of Radionuclides in the Unsaturated Zone

Transport of radionuclides in the unsaturated zone depends to some extent on the location from which they are released. In the northern area of the repository, water is expected to move principally within fractures. Average travel times for nonsorbing solutes from the repository to the saturated zone from the northern area of the repository are on the order of 5 to 100 years for an infiltration rate of 12 mm/yr [0.47 in/yr] (SAR Figure 2.3.8-36). Conversely, in the southern repository area, the Calico Hills nonwelded tuff unit has higher matrix permeability that can accommodate flow almost entirely within the rock matrix (porous flow). Average travel times from the southern repository area to the saturated zone are on the order of 500 to 5,000 years for an average infiltration rate of 12 mm/yr [0.47 in/yr] (SAR Figure 2.3.8-36).

For sorbing radionuclides, travel times depend on the radionuclide-specific sorption coefficient. More strongly sorbing aqueous species, such as Pu-242, have transport times on the order of hundreds of thousands of years and longer in the southern area. Radionuclide species with half-lives that are short relative to their unsaturated zone transport times can be significantly or completely decayed before reaching the water table (e.g., Cs-137, Sr-90, Ra-226, Th-229, Th-230, Th-232, Pu-238, Am-241, Am-243).

Table 17-4 shows how the combined processes affect flow of different radionuclides through the unsaturated zone by providing representative transport times for nonsorbing Tc-99, moderately sorbing Np-237, strongly sorbing Pu-242, and Pu-242 attached to colloids.

Transport of Radionuclides in the Saturated Zone

Radionuclides released from a Yucca Mountain repository would eventually enter the saturated zone within the fractured volcanic tuffs of the Crater Flat group. Transport away from the repository area would occur through permeable flowing fracture networks in the volcanic aquifer system for more than 10 km [6.2 mi] and transition to a valley fill alluvial flow system for the last few kilometers before reaching the accessible environment approximately 18 km [11.2 mi] from the southern boundary of the repository footprint. The exact location of the volcanic rock–alluvium contact is uncertain and is treated stochastically in the saturated zone transport abstraction model using an alluvium uncertainty zone. The fracture flow path for the

Table 17-4. Radionuclide Transport Times in the Unsaturated Zone for the Northern and Southern Repository Areas From DOE Breakthrough Curves		
	Transport Time* for Release in Northern Repository Area	**Transport Time* for Release in Southern Repository Area**
Tc-99	10 years	1,000 years
Np-237	10 years	10,000 years
Pu-242	30 years	>1 million years
Pu-242 irreversible colloids	100 years	1,000 years

*Transport times reflect approximate arrival for 50 percent of peak concentration for a model case with point releases at representative locations in northern and southern model areas, representative parameter values, and Glacial Transition 10th percentile infiltration map. See SAR Figures 2.3.8-43 for Tc-99, 2.3.8-44(b) for Np-237, 2.3.8-47(a) for Pu-242, and 2.3.8-48 for Pu-242 irreversible colloids for complete breakthrough curves for all radionuclides.

volcanic tuff is conceptualized as being relatively fast because the effective porosity is relatively small [average estimated value of 0.001 (SAR Table 2.3.9-4)]. Flow in the alluvial portion of the flow system is conceptualized as relatively slow because the effective flow porosity is relatively high [average estimated value of 0.18 (SAR Table 2.3.9-4)]. Overall, the transport time for nonsorbing radionuclides ranges from about 10 years to several thousand years (SAR p. 2.3.9-9). Sorbing radionuclides can be significantly delayed by sorption to alluvium mineral grains in which case transport times for strongly sorbing radionuclides generally exceed 10,000 years (SAR p. 2.3.9-9). Table 17-5 provides transport times for select radionuclides representing a range of sorption behavior taken from BSC Table 6-10a (2005ak).

Table 17-5. Summary of DOE Simulated Transport Times in the Saturated Zone Under Glacial-Transition Climate State		
Species	**Range of Median Transport Times (years)**	**Median Transport Time Among All Realizations**
C-14, Tc-99, I-129 (aqueous, nonsorbing)	10 to 22,190	230
Reversible colloids: americium, thorium	1,000 to >1 million	>1 million
Reversible colloids: plutonium	3,000 to >1 million	95,000
Neptunium	100 to 455,300	3,700
Irreversible colloids: plutonium, americium	100 to 501,900	4,500

2.2.1.4.1.3.3.1.1.6 Summary of DOE's TSPA for Annual Dose to the Reasonably Maximally Exposed Individual (RMEI)

Following postclosure engineered barrier system (EBS) release and groundwater radionuclide transport, the DOE TSPA model executes the biosphere model abstraction to calculate biosphere radionuclide transport and the annual dose to the reasonably maximally exposed individual (RMEI). The exposure scenarios implemented in the DOE TSPA model (i.e., groundwater, volcanic ash) calculate annual dose to an individual adult member of a hypothetical farming community located 18 km [11.2 mi] south of the potential repository along the path of groundwater flow. Exposure pathways in the DOE biosphere model are based on assumptions about residential and agricultural uses of the water and indoor and outdoor activities. These pathways include ingestion, inhalation, and direct exposure to radionuclides deposited to soil from irrigation (SAR Section 2.3.10.1). Ingestion pathways include drinking contaminated water, eating crops irrigated with contaminated water, eating food products produced from livestock raised on contaminated feed and water, eating farmed fish raised in contaminated water, and inadvertently ingesting soil. Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products resulting from the high-level radioactive waste.

DOE biosphere model results are quantified by the Biosphere Dose Conversion Factors (BDCFs). A biosphere dose conversion factor is the calculated annual dose to the reasonably maximally exposed individual (RMEI) from all potential exposure pathways as a result of a unit concentration of a radionuclide in groundwater or surface soil mixed with volcanic ash (SAR Section 2.3.10.1). Mean groundwater exposure scenario biosphere dose conversion factors and primary exposure pathways (from SAR Tables 2.3.10-11 and 2.3.10-12) for radionuclides that are important contributors to DOE's TSPA annual dose results (SAR Figure 2.4-26 a and b) are provided in Table 17-6. (Note: The volcanic ash exposure scenario for the igneous eruptive modeling case is discussed in the next section.)

The average annual doses are largest for the seismic ground motion and igneous intrusive modeling cases (generally a factor of 10 or more larger than the other modeling cases; see SAR Figure 2.4-18). Tc-99 (a nonsorbing radionuclide) is the largest contributor to the average

Table 17-6. DOE Groundwater BDCFs*

Radionuclide	Mean BDCF Sv/yr per Bq/m^3 [mrem/yr per pCi/L]	Primary Pathways
Tc-99	1.1×10^{-9} [4.1×10^{-3}]	42% drinking water 37% animal product 17% crop
Np-237	2.7×10^{-7} [1.0]	56% inhalation 29% drinking water
Pu-242	9.1×10^{-7} [3.4]	75% inhalation 19% drinking water

*Biosphere dose conversion factors

annual dose in the initial 10,000 years. Tc-99 accounts for approximately 0.001 mSv/yr [0.1 mrem/yr] of the peak of the overall average annual dose of approximately 0.003 mSv/yr [0.3 mrem/yr] (SAR Figure 2.4-20a). After 10,000 years and up to 1 million years, the peak of the overall average annual dose occurs at 1 million years, with Pu-242 and Np-237 being the largest contributors to the peak of the overall average annual dose. Pu-242 and Np-237 account for approximately 0.01 mSv/yr [1.0 mrem/yr] of the peak of the overall average annual dose of approximately 0.02 mSv/yr [2.0 mrem/yr] at 1 million years (SAR Figure 2.4-20b).

2.2.1.4.1.3.3.1.2 NRC Staff Evaluation of DOE's TSPA Calculation Related to Groundwater Releases

The NRC staff conducted confirmatory calculations to assist its review of DOE's TSPA results. The confirmatory calculations provide both a quantitative understanding of the attributes of the performance assessment and an understanding of whether there is a general consistency between submodels of the performance assessment and the overall results, including uncertainty (e.g., whether the timing and extent of breaching of the waste package are consistent with the timing and magnitude of the average annual dose). The confirmatory calculations were performed for selected time periods (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide perspective on the time-dependent nature of waste package failure, associated radioactive decay, and release of specific radionuclides. Detailed documentation of the NRC staff's confirmatory calculation is provided in NRC and CNWRA (2011aa).

The confirmatory calculations are based on the NRC staff's understanding of the TSPA calculation obtained from its SAR review, including the TSPA models and the code's intermediate outputs. Thus the confirmatory calculations address key quantitative attributes of the repository system to help evaluate overall performance. This approach provides a straightforward method for determining whether the TSPA results provide a credible representation of the repository performance (i.e., the average annual dose curve is consistent with the model abstractions, probabilities, and treatment of uncertainties, which have been reviewed in TER Sections 2.2.1.2–2.2.1.3.14).

In assessing the credibility of DOE's TSPA average annual dose curve resulting from the groundwater pathway, separate confirmatory calculations were conducted for (i) the amount of water entering failed waste packages, (ii) the release of radionuclides from the waste packages, (iii) transport of radionuclides through the saturated and unsaturated zones, and (iv) annual dose to the reasonably maximally exposed individual (RMEI).

Amount of Water Entering Failed Waste Packages

The NRC staff determined a representative quantity of seepage water entering the waste package using average values from the SAR for the seepage rates, seepage fraction. Specifically, the NRC staff divided the seepage rates by the seepage fraction (Table 17-2) to obtain the seepage rate at dripping locations in the drifts and multiplied this rate by the cross-sectional area of an emplacement drift {27.5 m^2 [296 ft^2]} to determine the seepage volume entering the drift. The amount of seepage water entering the drift that then enters the waste package is determined by the extent of the damage to the surface of the waste package (i.e., the size of the opening in the waste package from patch openings). DOE's TSPA assumes when 4 percent or more of the surface is failed, all seepage enters the waste package; otherwise, the amount of seepage entering the waste package linearly decreases below 4 percent to 0 percent. NRC's approach closely approximates the value DOE used (see TER

Section 2.2.1.3.3.3.3 for the NRC staff evaluation of the quantity of water entering the waste package). The average damage to the surface of the waste package was obtained from DOE Enclosure 1, Response Number 1, Figures 11, 12, 15, and 16 (2009bj). Table 17-7 presents the average volume of seepage that might enter a waste package that has patch failure (assuming seepage is present and the drip shield has failed) at specific times over the 1-million-year period (i.e., 10,000; 100,000; 400,000; and 800,000 years). One particularly important aspect of the values in Table 17-7 is that the average amount of seepage water entering patches, due to ruptures and punctures, decreases between 400,000 years and 800,000 years. This counterintuitive result is due to (i) ruptures, which can damage a larger surface area than punctures, occurring at early times and dominating the early average value for damage and (ii) punctures, occurring primarily at later times (when drifts are degraded and drip shields and waste packages are weakened by corrosion) and dominating the later average values. General corrosion of the waste package after very long times results in the largest amount of water contacting waste for the seismic ground motion modeling case because there are approximately 5 times more waste packages breached by general corrosion than by ruptures and punctures after 800,000 years (see Table 17-3). The igneous intrusion modeling case, which assumes the waste package and drip shield no longer prevent water from contacting waste, represents the maximum amount, on average, of water that can enter a failed waste package and contact waste.

Release of Radionuclides From Waste Packages

Release rates of radionuclides from an individual waste package are dependent primarily on the type of radionuclide. High mobility characterizes the soluble, nonsorbing radionuclides (e.g., Tc-99, I-129, Cl-36, Se-79) that result in relatively rapid waste package depletion after failure, whereby repository release is controlled by the amount present in the waste package or inventory and the degradation rate of the waste form. The overall release rate from the repository is dependent on the releases from the individual waste packages and how the releases from all the failed waste packages add together (or overlap at a particular time) to produce an overall release rate for the repository. For example, if all the packages failed at the same time, then the releases from all the waste packages would be occurring at the same time

Table 17-7. NRC Staff Confirmatory Calculation Results for the Volume of Seepage Water Entering Patch Failures in a Single Waste Package for Seismic Ground Motion (Ruptures, Punctures, and General Corrosion) and Igneous Intrusion Modeling Cases for CSNF* and CDSP† Waste Packages

	Volume (in liters/yr)			
Process	**10,000 Years**	**100,000 Years**	**400,000 Years**	**800,000 Years**
Ruptures and punctures	CSNF 0 CDSP 208	CSNF 0 CDSP 630	CSNF 63 CDSP 94	CSNF 47 CDSP 47
General corrosion	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 11 CDSP 13	CSNF 47 CDSP 63
Igneous intrusion	CSNF 609 CDSP 609	CSNF 889 CDSP 889	CSNF 889 CDSP 889	CSNF 889 CDSP 889

*Commercial spent nuclear fuel
†Codisposal
Note: Values of zero indicate no failed packages for the indicated failure type and time.

and would combine to produce the repository release rate. If, however, waste packages fail at different times, the potential for the releases to overlap in time will depend on the length of time between the failed packages and the time it takes a waste package to release the inventory of a particular radionuclide. When releases from a waste package are somewhat rapid, occurring over hundreds to thousands of years as is the case for the high-mobility radionuclides, the potential for releases from all the waste packages to overlap in time is reduced unless all the waste packages fail within the same time period over which the rapid release occurs. High release rates will persist for short periods of time (e.g., hundreds to thousands of years); thus, the overlap period for high waste package release rates will be short (a smaller number of waste package releases could potentially overlap in time). In contrast, low release rates may persist for hundreds of thousands of years and longer, and the overlap time period would be much longer and include the potential for a larger number of failed packages to contribute to the overall repository release rate.

As part of the NRC staff confirmatory calculation (NRC and CNWRA, 2011aa), a simplified approach was used to estimate releases from the waste package for three radionuclides important to the annual dose: Tc-99 (a soluble, nonsorbing radionuclide) and Np-237 and Pu-242 (relatively insoluble, sorbing radionuclides). The release rate from the repository engineered barriers is determined by multiplying the release rate from a single waste package times the number of waste package failures. This simple approach has the potential to calculate releases from the repository that would exceed, over the time period of the waste package failures, the amount of material that was present in the failed waste packages when releases rates are high. A bounding value for the release from a single waste package was calculated to ensure the releases from the repository would not exceed the amount of material present in the failed waste packages [bounding value is calculated as the inventory of a single waste package times the waste package failure rate; see NRC and CNWRA (2011aa) for further details]. The release rate from the engineered barriers for the repository were determined for selected time periods (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide some perspective on the time-dependent nature of waste package failure, the effect of radioactive decay, and release rate for a specific radionuclide.

For soluble, nonsorbing radionuclides (e.g., Tc-99), the release rate from the waste packages is assumed to be relatively rapid including release by diffusion through small cracks. Because diffusional release of soluble, nonsorbing radionuclides can be significant, all waste package breach types (e.g., rupture and puncture, general corrosion patches, stress corrosion cracks) are significant to estimating releases from the repository for the soluble nonsorbing radionuclides. Consistent with a rapid release time for soluble, nonsorbing radionuclides, DOE has stated that the release of Tc-99 tracks the waste package failure (SAR p. 2.4-63). The NRC staff used the bounding value for the release rate of Tc-99 in its confirmatory calculation because a relatively rapid release rate for a soluble, nonsorbing radionuclide can result in most, if not all, of the inventory being released over time periods that are short (e.g., thousands of years) relative to the time periods of the confirmatory calculation. The results of this simple, bounding calculation for Tc-99 showing the peak values at various times for the seismic ground motion modeling case and the igneous intrusion modeling case appear in Table 17-8. (Note: The waste package failure rate for the intrusion modeling case also includes a multiplicative factor to account for the annual probability for the intrusive event to occur.) Given that the majority of waste package failures in the seismic ground motion modeling case are due to crack failures (see Table 17-3) where the only release will be from the diffusion process, the diffusional releases are significant for the release of soluble, nonsorbing radionuclides (e.g., Tc-99) in the ground motion modeling case.

Table 17-8. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Tc-99 (Seismic Ground Motion Modeling Case) for CSNF* and CDSP† Waste Packages				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Waste package failure rate (waste packages/yr)‡	CSNF 1.6×10^{-4} CDSP 3.4×10^{-3}	CSNF 2.1×10^{-4} CDSP 0.011	CSNF 2.4×10^{-3} CDSP 1.1×10^{-3}	CSNF 7.0×10^{-3} CDSP 1.7×10^{-3}
Inventory§ (grams)	CSNF 7,405 CDSP 1,131	CSNF 5,526 CDSP 844	CSNF 2,082 CDSP 318	CSNF 567 CDSP 86
Average release rate (grams/yr)	CSNF 1.2 CDSP 3.8	CSNF 1.2 CDSP 9.3	CSNF 5.0 CDSP 0.35	CSNF 4.0 CDSP 0.15

*Commercial spent nuclear fuel
†Codisposal
‡Determined using all waste package failures types (i.e., cracks and patches) from Table 17-3 for the time periods (i.e., 0 to 10,000 years; 10,000 to 100,000 years; 100,000 to 400,000 years; and 400,000 to 800,000 years).
§Inventory is for one waste package at the specific time (Tc-99 half-life is 213,000 years); initial inventory from SAR Volume 2, Table 2.3.7-5.

Much lower mobility characterizes the relatively insoluble, sorbing nuclides (e.g., Np-237, Pu-242). Advective releases of these radionuclides are limited by maximum limits on their concentrations in water (from either precipitation or sorption to stationary corrosion products) and water flow through the waste package. Because releases of the lower mobility nuclides can take tens of thousands of years and longer, the potential for releases from individual waste packages to overlap in time is greater. For the concentration-limited radionuclides (e.g., Pu-242 and Np-237), DOE has explained that releases will be significantly lower than the release rates for soluble radionuclides (e.g., Tc-99) and will increase as water flow into the waste packages increases (e.g., as corrosion patch area increases in size over time more water may enter the waste package; see SAR p. 2.4-63).

For the relatively insoluble, sorbing radionuclides (e.g., Np-237, Pu-242), the approximation for the repositorywide engineered barrier system (EBS) release rate accounts for key aspects of repository performance that affect the release rate from the waste package (i.e., solubility limits, volume of water flux moving through a breached waste package, corrosion products, and radioactive decay and production). The release rate from the repository engineered barriers is determined by multiplying the release rate from a single waste package times the number of waste package failures, subject to the previously described constraint that the releases from the repository cannot exceed the amount of material present in the failed waste packages. Generally, the constraint on the release rate is necessary when the inventory is small and/or when the solubility limits and water flow rates are high. As the length of time for the inventory of a specific radionuclide to be completely released from a waste package increases, the potential for releases from waste packages, which failed at different times, to overlap or combine and result in larger releases from the repository increases. For example, if the time to release the entire inventory of a given radionuclide was 100,000 years, then the releases from all waste packages that failed within 100,000 years would all overlap by the end of the 100,000-year period. The NRC staff confirmatory calculation accounted for the overlap of waste package releases, which can vary with time due to radioactive decay and increasing water flow through the waste package, attributed to an increased damaged area of waste package patch breaches, which generally occurs at later times [e.g., increased general corrosion patch area at longer times, as shown in DOE Enclosure 1, Response Number 1, Figures 11 and 12 (2009bj)].

Generally, for the insoluble radionuclides, the larger repository release rates occur at later times (e.g., hundreds of thousands of years) when a larger number of waste packages are breached by patch failures, which allow seepage water to enter the waste package and release radionuclides by the advective flow of water out of the waste package. The advective releases, when present, tend to be significantly larger than the diffusive releases [see NRC staff analysis in NRC and CNWRA (2011aa)]. Therefore, the NRC staff's confirmatory calculation considers only advective releases out of the waste package for the insoluble radionuclides, which occurs only for failed packages in the dripping environment after the drip shield fails. The NRC staff's calculation assumes that the drip shield is always failed when the waste package is breached by a patch failure (i.e., an opening sufficiently large that dripping water can enter the waste package and the waste form degradation rate does not limit the release rate). Advective releases are calculated by estimating (i) the failure rate for waste packages breached by patches, such as by general corrosion and ruptures and punctures from the seismic ground motion modeling case; (ii) the amount of seepage that enters and flows through the waste package, which is dependent on the size of the waste package holes or patches, the seepage fraction and seepage rates, and for the NRC staff's calculation, an assumption that the drip shield is not functioning; and (iii) the effects of solubility limits and corrosion products on the concentration of radionuclides in the water flowing through the waste package. The results of the NRC staff's simple calculations for Np-237 and Pu-242 for the ground motion modeling case are shown Table 17-9. To account for uncertainty in the release of these radionuclides, solubility limits were based on the upper and lower values representative of the solubility limit

Table 17-9. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Np-237 and Pu-242 in the Seismic Ground Motion Modeling Case for CSNF* and CDSP† Waste Packages

	10,000 Years	100,000 Years	400,000 Years	800,000 Years
General corrosion failure rate‡ (waste packages/yr)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 1.0×10^{-6} CDSP 1.6×10^{-5}	CSNF 8.2×10^{-4} CDSP 5.9×10^{-4}
Water flux through general corrosion§ (liters/yr/waste package)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 11 CDSP 13	CSNF 47 CDSP 63
Rupture and puncture failure rate‡ (waste packages/yr)	CSNF 0 CDSP 3.0×10^{-5}	CSNF 0 CDSP 4.4×10^{-6}	CSNF 8.2×10^{-5} CDSP 4.3×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.0×10^{-5}
Water flux through ruptures and punctures (liters/yr/waste package)	CSNF 0 CDSP 208	CSNF 0 CDSP 630	CSNF 63 CDSP 94	CSNF 47 CDSP 47
Np-237 inventory‖ (grams/waste package)	CSNF 15,530 CDSP 504	CSNF 15,084 CDSP 490	CSNF 13,687 CDSP 444	CSNF 12,024 CDSP 390

Table 17-9. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Np-237 and Pu-242 in the Seismic Ground Motion Modeling Case for CSNF* and CDSP† Waste Packages (continued)				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Np-237 release rate¶ (grams/yr)	CSNF 0 CDSP 2.9×10^{-4} to 4.7×10^{-3}	CSNF 0 CDSP 1.5×10^{-3}	CSNF 0.016 to 0.32 CDSP 0.014 to 0.018	CSNF 0.20 to 4.0 CDSP 0.17
Pu-242 inventory‖ (grams/waste package)	CSNF 5,360 CDSP 39	CSNF 4,542 CDSP 33	CSNF 2,614 CDSP 19	CSNF 1,251 CDSP 9
Pu-242 release rate¶ (grams/yr)	CSNF 0 CDSP 8.1×10^{-5} to 3.6×10^{-4}	CSNF 0 CDSP 1.0×10^{-4}	CSNF 4.5×10^{-3} to 0.15 CDSP 7.4×10^{-4} to 7.6×10^{-4}	CSNF 0.056 to 0.83 CDSP 4.0×10^{-3}

*Commercial spent nuclear fuel
†Codisposal
‡Determined from Table 17-3 patch failures (i.e., ruptures, punctures, and general corrosion) for the time periods (i.e., 0 to 10,000 years; 10,000 to 100,000 years; 100,000 to 400,000 years; and 400,000 to 800,000 years).
§Water flux taken from Table 17-7.
‖Inventory is for one waste package at the specific time (Np-237 half-life is 2.14 million years, Pu-242 half-life is 376,300 years); initial inventory from SAR Volume 2, Table 2.3.7-5 (Np-237 inventory includes complete decay of Am-241 into Np-237).
¶Release rate determined with solubility limits of 0.3 to 6 mg/L for Np-237 and 0.006 to 0.5 mg/L for Pu-242 and corrosion product factor of 0.05 for Np-237 and 0.7 for Pu-242.

range DOE used in its TSPA analysis (see TER Section 2.2.1.3.4). The release rates from the commercial spent nuclear fuel waste packages overall are larger than the releases from the codisposal fuel packages due to the larger inventory for these radionuclides present in commercial spent nuclear fuel and the larger number of commercial spent nuclear fuel waste packages in the repository. The NRC staff's calculation does not account for releases of radionuclides from the waste packages that are associated with colloids. This is because releases of radionuclides that are not associated with colloids (i.e., radionuclides that are dissolved in water) are larger than releases of radionuclides associated with colloids (see NRC staff's review in TER Section 2.2.1.3.4.3.4).

The NRC staff performed confirmatory calculations for the igneous intrusive modeling case similar to those performed for the seismic ground motion modeling case. The igneous intrusive modeling case is somewhat simpler in that it is assumed that all waste packages are completely failed (i.e., no diversion of seepage water) once the event occurs and the seepage fraction is 100 percent (i.e., all packages experience dripping water). The NRC staff accounted for the probability of the event occurring (i.e., mean annual frequency of an igneous event intersecting the repository is 1.7×10^{-8} per year; SAR p. 2.3.11-9) by incorporating the event probability into the determination of the number of waste package failures (see Table 17-3). The number of waste package failures in the seismic ground motion modeling case reported in Table 17-3 also represents values that incorporate the probability of the seismic events occurring. The releases for the soluble radionuclide Tc-99 and for the insoluble radionuclides Np-237 and Pu-242 from the igneous intrusive modeling case, as determined through an NRC staff confirmatory calculation, are presented in Tables 17-10 and 17-11, respectively.

Transport of Radionuclides through the Unsaturated and Saturated Zones

As part of the NRC staff's confirmatory calculation, the NRC staff developed multiplicative factors to account for the effect of transport in DOE's TSPA evaluation for the unsaturated and saturated zones on the releases of radionuclides to the reasonably maximally exposed individual (RMEI) location. The effectiveness of the travel times in the geosphere is related to the time at which the annual dose occurs. For example, a delay of tens of thousands years is not expected to significantly affect annual dose at hundreds of thousands of years but would affect annual dose at 10,000 years. On the basis of these considerations, the NRC staff developed factors for representing the effects of the unsaturated and saturated zones for reducing radionuclide releases (see Table 17-12). The nonsorbing radionuclide Tc-99 is typified by short delay time, and thus the releases are unaffected (i.e., no reduction in the release rates) by either the unsaturated or saturated zones. As noted in Table 17-4, DOE's representation for the unsaturated zone is split with slower releases for sorbing radionuclides for half of the repository, which would be in the southern portion. For the NRC staff's confirmatory calculation, the unsaturated zone reduces the releases for the sorbed radionuclides (Np-237 and Pu-242) up to 50 percent due to delay times on the order of thousands of years and longer in the southern portion of the unsaturated zone. Sorption of radionuclides causing the delay or slowing of radionuclide travel times for thousands of years will be less noticeable at longer times (e.g., 100,000 years). After 10,000 years, only the more strongly sorbed radionuclide Pu-242 continues to be reduced by the effectiveness of the unsaturated zone. (Note: a small quantity of Pu-242 is associated with irreversible colloids that would not be reduced by

Table 17-10. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Tc-99 in the Igneous Intrusive Modeling Case for CSNF* and CDSP† Waste Packages

	10,000 Years	100,000 Years	400,000 Years	800,000 Years
Waste package failure rate‡ (waste packages/yr)	CSNF 1.4×10^{-4} CDSP 6.0×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}
Inventory§ (grams)	CSNF 7,405 CDSP 1,131	CSNF 5,526 CDSP 844	CSNF 2,082 CDSP 318	CSNF 567 CDSP 86
Average release rate (grams/yr)	CSNF 1.0 CDSP 0.068	CSNF 0.77 CDSP 0.049	CSNF 0.29 CDSP 0.018	CSNF 0.079 CDSP 5.0×10^{-3}

*Commercial Spent Nuclear Fuel
†Codisposal
‡Determined using values in Table 17-3 for the time periods (i.e., 0 to 10,000 years; 10,000 to 100,000 years; 100,000 to 400,000 years; and 400,000 to 800,000 years).
§Inventory is for one waste package at the specific time (Tc-99 half-life is 213,000 years); initial inventory from SAR Volume 2, Table 2.3.7-5.

Table 17-11. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Np-237 and Pu-242 in the Igneous Intrusive Ground Motion Case for CSNF* and CDSP† Waste Packages

	10,000 Years	100,000 Years	400,000 Years	800,000 Years
Waste package failure rate‡ (packages/yr)	CSNF 1.4×10^{-4} CDSP 6.0×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}
Water flux§ (liters/yr/waste package)	CSNF 609 CDSP 609	CSNF 889 CDSP 889	CSNF 889 CDSP 889	CSNF 889 CDSP 889
Np-237 inventory‖ (grams/waste package)	CSNF 15,530 CDSP 504	CSNF 15,084 CDSP 490	CSNF 13,687 CDSP 444	CSNF 12,024 CDSP 390
Np-237 release rate¶ (grams/yr)	CSNF 0.013 to 0.26 CDSP 5.5×10^{-3} to 0.030	CSNF 0.19 to 2.1 CDSP 0.028	CSNF 0.75 to 1.9 CDSP 0.026	CSNF 1.5 to 1.7 CDSP 0.023
Pu-242 inventory‖ (grams/waste package)	CSNF 5,360 CDSP 39	CSNF 4,542 CDSP 33	CSNF 2,614 CDSP 19	CSNF 1,251 CDSP 9
Pu-242 release rate¶ (grams/yr)	CSNF 3.6×10^{-3} to 0.30 CDSP 1.5×10^{-3} to 2.3×10^{-3}	CSNF 0.052 to 0.64 CDSP 1.9×10^{-3}	CSNF 0.21 to 0.37 CDSP 1.1×10^{-3}	CSNF 0.18 CDSP 5.2×10^{-4}

*Commercial Spent Nuclear Fuel
†Codisposal
‡Determined from Table 17-3 for the time periods (i.e., 0 to 10,000 years; 10,000 to 100,000 years; 100,000 to 400,000 years; and 400,000 to 800,000 years).
§Water flux taken from Table 17-7.
‖Inventory is for one waste package at the specific time (Np-237 half-life is 2.14 million years, Pu-242 half-life is 376,300 years); initial inventory from SAR Volume 2, Table 2.3.7-5 (Np-237 inventory includes complete decay of Am-241 into Np-237).
¶Release rate determined with solubility limits of 0.3 to 6 mg/L for Np-237 and 0.006 to 0.5 mg/L for Pu-242 and corrosion product factor of 0.05 for Np-237 and 0.7 for Pu-242.

sorption; however, this amount is small in DOE's TSPA analysis, as explained in TER Section 2.2.1.4.1.3.3.1.1.4, and is not considered in the NRC staff's confirmatory calculation.) The saturated zone provides somewhat more effectiveness for Pu-242 (reversible colloids and dissolved Pu-242) at longer times and is assumed to significantly reduce releases (i.e., 97 percent reduction) over the 10,000-year time period due to the median

Table 17-12. NRC Staff Confirmatory Calculation Values for the Effectiveness (Expressed as a Percentage Reduction in Release) of the Unsaturated and Saturated Zones for Reducing Release Rates for Specific Radionuclides				
	0 to 10,000 Years	**10,000 to 200,000 Years**	**200,000 to 600,000 Years**	**600,000 to 1 Million Years**
Tc-99 (nonsorbing)	UZ 0% SZ 0%	UZ 0% SZ 0%	UZ 0% SZ 0%	UZ 0% SZ 0%
Np-237 (moderately sorbing)	UZ 45% SZ 40%	UZ 5% SZ 10%	UZ 0 % SZ 5%	UZ 0% SZ 0%
Pu-242 (aqueous) (strongly sorbing)	UZ 50% SZ 97%	UZ 50% SZ 60%	UZ 50% SZ 35%	UZ 50% SZ 25%

transport time of 95,000 years (see Table 17-5). The saturated zone is less effective for Np-237 because median transport time is 3,700 years (see Table 17-5). The confirmatory calculation (i) considered radionuclides that are released to the reasonably maximally exposed individual (RMEI) location representing the more significant contributors to the TSPA calculated average annual dose and (ii) did not consider releases of a variety of other radionuclides that were reduced to much lower levels due to radioactive decay when the waste package is intact and during transport in the unsaturated and saturated zones (e.g., reversible colloids for americium and thorium are delayed more than 1 million years in the saturated zone; see Table 17-5), because the contributions to the average annual dose would have been so small.

Annual Dose to the Reasonably Maximally Exposed Individual (RMEI)

The NRC staff's confirmatory calculation for the annual dose to the reasonably maximally exposed individual (RMEI) is completed by multiplying the biosphere dose conversion factors (BDCFs) in Table 17-6 with the NRC staff's estimated saturated zone releases to estimate an annual dose for comparison with DOE's TSPA results. Tables 17-13 through 17-15 compare the confirmatory calculation and the TSPA results. Overall, the annual doses from the confirmatory calculation are in general agreement with the TSPA results (e.g., a majority of either single value comparisons are within a factor of two or single values for the TSPA results fall between the upper and lower values when the confirmatory calculation provides both an upper and lower value). The igneous intrusive modeling case, which is already somewhat simplified in the TSPA model by assuming all waste packages fail when the event occurs, tends to exhibit the best fit between the confirmatory calculation and the TSPA results. The fit for Tc-99 also exhibits a better fit regardless of the modeling case because the representation of Tc-99 in the repository is less complex: high solubility and mobility for Tc-99 limits the factors affecting release and transport of Tc-99. Although the ground motion modeling case is a bit more complicated due to the variety and timing of waste package breaches (e.g., cracks, ruptures, and patches), the results of the NRC staff's confirmatory calculation are in general agreement with the TSPA results. There is no precise agreement between the NRC staff's confirmatory calculation and the results of DOE's Total System Performance Assessment (TSPA) results due to the simplifying assumptions made in the confirmatory calculation [see NRC and CNWRA (2011aa) for further details on assumptions for the NRC staff confirmatory

Table 17-13. NRC Staff Confirmatory Calculation Results for the Average Dose Estimates for Tc-99 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/year) (from Tables 17-8 and 17-10)	Seismic 5.0 Igneous 1.1	Seismic 10 Igneous 0.82	Seismic 5.4 Igneous 0.31	Seismic 4.2 Igneous 0.084
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	1	1	1	1
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 0.094 Igneous 0.021	Seismic 0.19 Igneous 0.015	Seismic 0.10 Igneous 5.8×10^{-3}	Seismic 0.079 Igneous 1.6×10^{-3}
DOE's TSPA average annual dose‡ (mrem/year)	Seismic 0.10 Igneous 0.017	Seismic 0.16 Igneous 0.013	Seismic 0.13 Igneous 7.0×10^{-3}	Seismic 0.090 Igneous 1.6×10^{-3}

*Biosphere dose conversion factor
†Annual dose calculation based on an annual water demand of 3,000 acre-ft per 10 CFR 63.112(c).
‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.

calculation]. The NRC staff's confirmatory calculation was used to confirm NRC's understanding of the key attributes of the repository performance in DOE's TSPA analyses and to assess that those attributes are consistent with DOE's dose results. The confirmatory calculation considered the effect on dose by (i) the number of waste packages and the extent of waste package damage, (ii) the drift seepage, (iii) solubility limits for individual radionuclides (including the effect of corrosion products), (iv) the inventory of specific radionuclides, (v) sorption in the geosphere, (vi) and the probability of disruptive events. Consistency between the confirmatory calculation and DOE's TSPA results provides further confidence that DOE's TSPA analysis is consistent with the model abstractions described in the SAR and reviewed in TER Sections 2.2.1.3.1 through 2.2.1.3.14.

Table 17-14. NRC Staff Confirmatory Calculation Results for the Annual Dose for Np-237 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case

	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/yr) (from Tables 17-9 and 17-11)	Seismic 2.9×10^{-4} to 4.7×10^{-3} Igneous 0.018 to 0.29	Seismic 1.5×10^{-3} Igneous 0.22 to 2.1	Seismic 0.030 to 0.34 Igneous 0.78 to 1.9	Seismic 0.37 to 4.2 Igneous 1.5 to 1.7
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	0.33	0.855	0.95	1
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	1	1	1	1
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 1.8×10^{-5} to 3.0×10^{-4} Igneous 1.0×10^{-3} to 0.018	Seismic 2.5×10^{-4} Igneous 0.036 to 0.34	Seismic 5.3×10^{-3} to 0.061 Igneous 0.14 to 0.34	Seismic 0.070 to 0.80 Igneous 0.29 to 0.32
DOE's TSPA average annual dose‡ (mrem/yr)	Seismic 1.5×10^{-6} Igneous 3.0×10^{-3}	Seismic 2×10^{-4} Igneous 0.05	Seismic 2×10^{-3} Igneous 0.13	Seismic 0.04 Igneous 0.22

*Biosphere dose conversion factor
†Annual dose calculation based on an annual water demand of 3,000 acre-ft per 10 CFR 63.112(c).
‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.

Table 17-15. NRC Staff Confirmatory Calculation Results for the Annual Dose for Pu-242 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/yr) (from Tables 17-9 and 17-11)	Seismic 8.1×10^{-5} to 3.6×10^{-4} Igneous 5.1×10^{-3} to 0.3	Seismic 1.0×10^{-4} Igneous 0.054 to 0.64	Seismic 5.2×10^{-3} to 0.15 Igneous 0.21 to 0.37	Seismic 0.060 to 0.83 Igneous 0.18
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	0.015	0.20	0.325	0.375
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	3.4	3.4	3.4	3.4
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 4.3×10^{-6} to 2.0×10^{-5} Igneous 2.8×10^{-4} to 0.016	Seismic 7.2×10^{-5} Igneous 0.040 to 0.47	Seismic 6.2×10^{-3} to 0.18 Igneous 0.25 to 0.44	Seismic 0.080 to 1.1 Igneous 0.25
DOE's TSPA average annual dose‡ (mrem/year)	Seismic 0 Igneous 0	Seismic 4.0×10^{-4} Igneous 0.05	Seismic 0.013 Igneous 0.23	Seismic 0.15 Igneous 0.23

*Biosphere dose conversation factor
†Annual dose calculation based on an annual water demand of 3,000 acre-ft per 10 CFR 63.112(c).
‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.

The NRC staff conducted its confirmatory calculation to assist in its review of DOE's TSPA models and calculations. The confirmatory calculation provides both a quantitative understanding of the attributes of the performance assessment and an understanding of whether there is a general consistency between submodels of the performance assessment and the overall results, including uncertainty (e.g., whether the timing and extent of breaching of the waste package is consistent with the timing and magnitude of the average annual dose). On the basis of its confirmatory calculation, the NRC staff makes the following observations:

- The NRC staff's confirmatory calculation follows the water and key radionuclides through the repository system (seepage to release to transport to annual dose) for the dominant modeling cases, whereby seismic ground motion and igneous intrusion events have the largest number of waste package failures and are the largest contributors to annual dose. The NRC staff notes that DOE's TSPA calculated average annual dose curve is reasonable in that the average annual doses are consistent with the intermediate model results [e.g., seepage, waste package failures, release of radionuclides from the engineered barrier system (EBS), and transport in the geosphere]. The NRC staff's review of the intermediate models of DOE's TSPA is described under the NRC staff's model abstraction review in TER Sections 2.2.1.3.1 through 2.2.1.3.14.

- DOE's description of the barriers important to waste isolation is fully consistent with the key attributes of the repository in the NRC staff's confirmatory calculation. In particular,

  (i) The waste package prevents significant releases for long periods of time (e.g., less than 1 percent of commercial spent nuclear fuel waste packages are breached after 100,000 years; see Table 17-3)

  (ii) Once breached, the releases from the waste package are limited due to the manner in which the waste package fails (i.e., majority of waste package breaches are due to cracks that do not allow seepage water to enter the waste packages), which limits the amount of water that enters the waste package; solubility limits and corrosion products within the waste package that reduce the release of many radionuclides from the waste package; and the limited amount of seepage that is present due to the upper natural barrier (i.e., rock layers above the repository)

  (iii) After release from the waste package, a variety of radionuclides sorb onto rock surfaces and are delayed for thousands of years in portions of both the unsaturated zone (see Table 17-9) and the saturated zone (see Table 17-10)

- The confirmatory calculation focused on certain long-lived radionuclides, on the order of 100,000 years or longer, that could eventually be released at the RMEI location in sufficient quantities to be important for dose calculations. Both a nonsorbing radionuclide (Tc-99) and sorbing radionuclides (Np-237 and Pu-242) were considered in the NRC staff's calculations. The initial inventory of high-level waste is composed of a large quantity (in terms of curies) of radionuclides (e.g., Cs-137, Sr-90, Am-241) with shorter half-lives, on the order of 1,000 years and less. For the short-lived radionuclides, DOE described in DOE Enclosure 7, Response Number 7, Table 1-3 (2009an) that delay in the geosphere has the capability to reduce releases nearly 100 percent.

- The effect of uncertainties on DOE's average annual dose curve is limited primarily because a number of important aspects of repository performance are near maximum values. For example, after 10,000 years nearly all the waste packages are dripped on (seepage fraction of 69 percent for seismic ground motion modeling case and 100 percent for igneous intrusion; see Table 17-2) and, given the 1-million-year period, a variety of long-lived radionuclides can eventually make it to the reasonably maximally exposed individual (RMEI) location (see Tables 17-4 and 17-5). Releases from the waste package will be affected by the failure rate for the waste package, including the areal extent of the waste package breaches, solubility limits, and the effect of corrosion products. The confirmatory calculation considered the low and high values of the solubility limits to provide some insight on how uncertainty in release from the waste package might impact the annual dose. Use of the highest solubility limit, as expected, increases the annual dose in the NRC staff confirmatory calculation. The estimated peak dose in the NRC staff's confirmatory calculation at 10,000 years is 0.0015 mSv/yr [0.15 mrem/yr] and at 800,000 years is 0.026 mSv/yr [2.6 mrem/yr].

On the basis of DOE's SAR description of the TSPA models and its results, the NRC staff's confirmatory calculation provides further confidence that DOE's average annual dose curve is consistent with the model abstractions, scenario probabilities, and the capabilities of the barriers important to waste isolation.

2.2.1.4.1.3.3.2 DOE's TSPA Calculation for the Volcanic Eruption Modeling Case

2.2.1.4.1.3.3.2.1 Summary of DOE Approach

An eruptive volcanic event at the repository involves the intersection of ascending magma and a drift and eruption at the surface (see TER Section 2.2.1.3.10). Radioactive material entrained in tephra can be transported downwind and deposited on the ground surface where potential exposures can occur from (i) inhalation of radionuclides due to high-level waste entrained in ash particles, which are suspended in the air, including the breathing of radon gas and its daughter products from high-level waste entrained in the ash deposited on the ground surface and (ii) ingestion of radionuclides from locally produced crops and animal products that are assumed to be contaminated from direct (e.g., crops grown in soil containing contaminated tephra) and indirect (e.g., animals raised on feed that has been grown in soil containing contaminated tephra) contact with contaminated tephra. Estimating the consequences of such an event is dependent on the concentration of radionuclides in tephra and the amount of ash persisting at the reasonably maximally exposed individual (RMEI) location (from both the direct deposition of tephra during the event and redistribution of tephra after the event due to water and wind action over time).

On average, the volcanic eruption modeling case impacts four (a range of one to seven) waste packages and entrains all of the waste into magma. Of this magma and waste, 30 percent is considered to form tephra; thus 30 percent of the waste in the waste packages hit by the event is, on average, contained in the tephra (range of 10 to 50 percent). Once radioactively contaminated volcanic tephra is present at the reasonably maximally exposed individual (RMEI) location, potential exposures are estimated for three specific pathways (external exposure; ingestion; and inhalation, which includes radon exposure) using Biosphere Dose Conversion Factors (BDCFs). DOE's BDCFs are provided in Tables 17-16 and 17-17 for some of the radionuclides included in DOE's Total System Performance Assessment (TSPA) relevant to the volcanic eruption modeling case.

The volcanic eruption modeling case average annual dose curve is one of the lowest dose curves of all the modeling cases resulting in average doses that are less than 10^{-5} mSv/yr [0.001 mrem/yr] at all times (SAR Figure 2.4-18). During the initial 10,000 years, the annual dose is dominated by Pu-239, Pu-240, and Am-241 (SAR Figure 2.4-32). For these three radionuclides, the inhalation exposure accounts for more than 98 percent of the average annual

Table 17-16. DOE Volcanic Eruption Modeling Case Short-Term and Long-Term Inhalation BDCFs*		
Radionuclide	**BDCF Sv/yr per Bq/kg [mrem/yr per pCi/g]**	**Primary Pathways**
Pu-239	4.0×10^{-7} [1.5] short term† 6.1×10^{-7} [2.3] long term	98% of Pu-239 eruptive dose is inhalation: 39% short term 60% long term
Am-241	3.2×10^{-7} [1.2] short term† 5.0×10^{-7} [1.8] long term	94% of Am-241 eruptive dose is inhalation: 37% short term 57% long term

*Biosphere dose conversion factors
†The short-term inhalation exposure is applicable only for the initial year of the eruption.

Table 17-17. DOE Volcanic Eruption Modeling Case Combined Ingestion, Radon, and External BDCFs*		
Radionuclide	**BDCF Sv/yr per Bq/m^2 [mrem/yr per pCi/m^2]**	**Primary Pathways**
Sr-90	1.8×10^{-9} [6.7×10^{-6}]	79% external exposure
Cs-137	7.2×10^{-9} [2.7×10^{-5}]	99% external exposure
Ra-226	3.3×10^{-8} [1.2×10^{-4}]	65% external exposure 33% radon decay products

*Biosphere dose conversion factors

dose for the volcanic eruption modeling case, as shown in SAR Table 2.3.10-15. At very early times (i.e., the initial 500 years), there is some contribution from Sr-90 and Cs-137 (primarily from external exposure). At very long times (i.e., after 100,000 years), the annual dose is dominated by Ra-226 (SAR Figure 2.4-32). These results are partially due to the half-lives for these radionuclides. Sr-90 and Cs-137 have half-lives less than 100 years, and Am-241 has a half-life of 432 years; thus the hazard is somewhat short lived. The longer term hazard is with Pu-239 (half-life of 24,000 years), Pu-240 (half-life of 24,000 years), and Ra-226, which is a daughter product in the long-lived U-234 chain (the half-life of U-234 is 240,000 years, whereas the half-life of Ra-226 is 1,600 years).

2.2.1.4.1.3.3.2.2 NRC Staff Evaluation of DOE's TSPA Calculation for the Volcanic Eruption Modeling Case

The NRC staff reviewed the TSPA and SAR with respect to representing an extrusive igneous event, including the treatment of uncertainty, and notes that DOE's approach is reasonable. In particular, the following is reasonable: (i) the probability for an extrusive event to intersect the repository and hit waste packages (TER Section 2.2.1.2.2), (ii) the model abstractions for disruption of the waste package by an extrusive igneous event (TER Section 2.2.1.3.10), (iii) the model abstractions for airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption following igneous disruption of waste packages and the redistribution of those radionuclides in soil (TER Section 2.2.1.3.13), and (iv) the volcanic exposure scenario for the reasonably maximally exposed individual (RMEI) (TER Section 2.2.1.3.14).

Generally, the largest contribution to annual dose is from the inhalation and external exposure pathways (see SAR Table 2.3.10-15). Annual dose from each of these pathways is dependent on the amount of radionuclides that are transported with the tephra (ash), and the inhalation pathway is also dependent on the amount of this material that is potentially available for inhalation by the reasonably maximally exposed individual (RMEI) (i.e., the mass loading of tephra and radioactive material in the air). In particular, the NRC staff notes that

- The amount of radionuclides entrained in tephra (ash) is reasonably conservative relative to entrainment of rock fragments in other volcanic conduits within similar and different geologic environments to Yucca Mountain (TER Section 2.2.1.3.10)

- Values for mass loading are appropriate and consistent with available information and are generally consistent with the NRC staff studies and analyses conducted over the past 5 years (TER Section 2.2.1.3.14)

The NRC staff also performed a confirmatory calculation to help evaluate the reasonableness of DOE's TSPA average annual dose curve for this modeling case [detailed documentation of the NRC staff's confirmatory calculation is provided in NRC and CNWRA (2011aa)]. Because the average annual dose for the volcanic eruption modeling case is so small {i.e., less than 10^{-5} mSv/yr [0.001 mrem/yr]}, a simplified calculation was performed for a few key radionuclides that would tend to bound some of the assumptions for this modeling case. The confirmatory calculation looked at radionuclides that include both the inhalation and direct exposure pathways and assumed that waste entrained in tephra (ash) was deposited directly at the reasonably maximally exposed individual (RMEI) location and persists without erosion removal at that location over the time period of the estimated dose (i.e., only radioactive decay reduces the level of radionuclides at the exposure location after the time of the initial deposition of the tephra). Additionally, the confirmatory calculation assumes tephra from the eruptive event is always deposited directly to the RMEI location (i.e., during the volcanic eruption the wind always blows in the direction of the RMEI such that ashfall occurs at the RMEI location). To provide a perspective on how radioactive decay and probability of the event affects the annual dose, two time periods were selected for the calculation: a time equal to the half-life of the radionuclide and three times the half-life. Radioactive decay will tend to decrease the annual dose to an individual over time. Tables 17-18 through 17-20 present the calculations for a few of the radionuclides that are most significant to performance or radiation exposure to the reasonably maximally exposed individual (RMEI) in the volcanic eruption modeling case.

As anticipated, the resulting annual doses from the NRC staff's confirmatory calculation are larger than the average annual doses calculated by DOE's detailed TSPA model. This is not

Table 17-18. NRC Staff Confirmatory Calculation Results for Pu-239 and Am-241 Annual Doses for the Volcanic Eruption Modeling Case (Inhalation Pathway)				
	Pu-239 (Half-Life 24,065 Years)		Am-241 (Half-Life 432 Years)	
Performance Aspect	**24,065 Yrs**	**72,195 Yrs**	**432 Yrs**	**1,296 Yrs**
Number of waste packages (SAR Figure 2.3.11-12)	4	4	4	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3	0.3	0.3	0.3
Tephra volume (km^3)	0.038	0.038	0.038	0.038
Inventory per CSNF* waste package (with mixed oxide added) [Ci] (SAR Table 2.3.7-5)	1,370	343	17,554	4,390
Concentration in ash (pCi/g)†	43	11	550	140
Weighted annual dose‡ (mrem/yr)	3.3×10^{-3}	2.6×10^{-3}	6.0×10^{-4}	4.6×10^{-4}
TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	8.0×10^{-5}	6.0×10^{-5}	3.0×10^{-5}	2.0×10^{-5}

*Commercial spent nuclear fuel waste package
†Concentration in ash calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
‡Weighted annual dose calculated by multiplying concentration with BDCF (Table 17-16), annual probability (1.4×10^{-9}), and the time.

unexpected because (i) the NRC staff's confirmatory calculation, which assumes direct deposits persist without erosion and the wind always blows in the direction of the RMEI, is more conservative than DOE's TSPA, which accounts for erosion of the tephra deposit, and (ii) the wind direction can vary (see TER Section 2.2.1.3.13). For the volcanic eruption modeling case (Tables 17-18 through 17-20), the annual dose from the confirmatory calculation represents the smallest dose for the modeling cases estimated in the NRC's confirmatory calculation {i.e., less than 0.0001 mSv [0.01 mrem]}. Consistency between the NRC staff's confirmatory calculation for the volcanic eruption modeling case (Tables 17-18 through 17-20) and the DOE's representation of the volcanic eruption modeling case provides further confidence that DOE's TSPA analysis is consistent with the assumptions and model abstractions described in the SAR and reviewed in TER Sections 2.2.1.2.2, 2.2.1.3.10, 2.2.1.3.13, and 2.2.1.3.14.

2.2.1.4.1.3.4 Statistical Stability of Average Annual Dose Estimates

2.2.1.4.1.3.4.1 Summary of DOE Approach

DOE addressed the question of the stability in SAR Section 2.4.2.2.2. The term *stability* refers to the numerical reproducibility of statistics (e.g., average annual dose) or their level of convergence as a function of model features such as size of the statistical sample and

Table 17-19. NRC Staff Confirmatory Calculation of Sr-90 and Cs-137 Annual Doses for the Volcanic Eruption Modeling Case (External Pathway)				
	Sr-90 (Half-Life 29 Years)		Cs-137 (Half-Life 30 Years)	
Performance Aspect	29 Yrs	87 Yrs	30 Yrs	90 Yrs
Number of waste packages (SAR Figure 2.3.11-12)	4	4	4	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3	0.3	0.3	0.3
Tephra volume (km^3)	0.038	0.038	0.038	0.038
Inventory per CSNF* waste package with mixed oxide added) [Ci] (SAR Table 2.3.7-5)	52,011	13,009	81,518	20,388
Ash areal concentration† (pCi/m^2)	1.6×10^7	4.1×10^6	2.6×10^7	6.4×10^6
Weighted annual dose‡ (mrem/yr)	4.4×10^{-6}	3.4×10^{-6}	2.9×10^{-5}	2.2×10^{-5}
TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	2.0×10^{-6}	1.8×10^{-6}	2.0×10^{-5}	1.0×10^{-5}

*Commercial spent nuclear fuel waste package
†Ash areal concentration calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density and assuming a 1-cm [0.39-in]-thick deposit (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
‡Weighted annual dose calculated by multiplying concentration with BDCF (Table 17-17), annual probability (1.4×10^{-9}), and the time.

numerical approximations. Variation in the TSPA results is a function of a particular combination of uncertain and variable parameters (DOE described its treatment of epistemic and aleatory uncertainty in SAR Section 2.4.2.1.1). DOE identified aleatory parameters as those parameters with uncertainty irreducible by additional experiments or site characterization. Examples of aleatory parameters are the time of seismic and igneous events, the extent of waste package damage during a seismic or faulting event, the location of the compromised or breached waste package in the repository, and the type of waste package [e.g., commercial spent nuclear fuel (CSNF) or codisposal (CDSP) waste packages] compromised after a disruptive event. The stability of the average annual dose will, in part, be a function of the size of the discrete sample of aleatory parameter values. DOE analyzed the effect of the size of these discrete samples by increasing the number of aleatory realizations from 30 to 90, considering more waste package damage fractions for the seismic and faulting modeling case, nd accounting for more event times (e.g., doubling the number of event times for the seismic ground motion modeling case, increasing the number of event times from 10 to 50 for the igneous intrusion modeling case) and determined that these types of changes would have a

Table 17-20. NRC Staff Confirmatory Calculation of Ra-226 Annual Dose for the Volcanic Eruption Modeling Case (External Pathway)	
Performance Aspect	**Ra-226 (Half-Life 1,600 Years) [U-238 (Half-Life 4.5 × 10^9 Years)] 1 Million Years***
Number of waste packages (SAR Figure 2.3.11-12)	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3
Tephra volume (km^3)	0.038
Inventory per CSNF† (waste package with mixed oxide added) [Ci] (SAR Table 2.3.7-5)	2.65
Ash areal concentration‡ (pCi/m^2)	840
Weighted annual dose§ (mrem/yr)	1.4×10^{-4}
TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	5.0×10^{-5}

*1-million-year period selected for this calculation due to persistence of Ra-226 based on decay of long-lived U-238. Ra-226 inventory assumes Ra-226 activity equals that of parent radionuclide U-238; SAR Table 2.3.7-5.
†Commercial spent nuclear fuel
‡Ash areal concentration calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density and assuming a 1-cm [0.39-in]-thick deposit (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
§Weighted annual dose calculated by multiplying concentration with BDCF (Table 17-17), annual probability (1.4×10^{-9}), and the time.

a minor effect on the magnitude of the overall average annual dose curve. DOE compared annual dose curves for a set of five realizations for all modeling cases in SAR Figures 2.4-55 to 61 and concluded, in qualitative terms, that the annual dose curve for the analyzed realizations was stable with respect to aleatory uncertainty.

DOE also examined the stability of the average annual dose curve to the treatment of the epistemic uncertainty. The epistemic parameters are generally those parameters that describe the repository components and behavior under nominal conditions (e.g., variability in hydraulic conductivity of a particular rock unit; variability in material properties over the surface of the waste package). DOE used a statistical sample size of 300 realizations for each modeling case in the SAR. To examine the stability of the annual dose curve with respect to the treatment of the epistemic uncertainty, DOE estimated dose statistics (mean, median, 5^{th} and 95^{th} percentiles) for the nominal modeling case considering 1,000 realizations and compared those statistics to corresponding 300-realization statistics (SAR Figure 2.4-38). DOE showed the 300-realization and 1,000-realization annual dose statistics (mean, median, 0.05 and 0.95 percentiles) were comparable (SAR Figure 2-4-38).

For all of the modeling cases, DOE considered three replicates with 300 realizations and compared statistics (mean, median, 0.05 and 0.95 percentiles) among the replicates (SAR Figures 2.4-37 to 2.4-52). Each replicate sample had the same number of realizations; however, the combination of sampled parameter values was different for each replicate. DOE qualitatively concluded that the statistics were similar for the three replicates. Also, DOE estimated 95 percent confidence bounds for the average annual dose using information from the replicates and a t-distribution with 2 degrees of freedom, as described in SNL Section J4.10 (2008ag). In all model cases, the 0.95 percentile in the average annual dose was relatively close {e.g., largest difference of 0.01 mSv/yr [1 mrem/yr] between the three replicates and generally much less for the vast majority of the 1-million-year period, as shown in SNL Figure J5-5(a) (2008ag)} to the overall average annual dose. DOE concluded the overall average annual dose, computed using 300 realizations, to be statistically stable, as described in SAR Section 2.4.2.2.2 and SNL Section 7.3.2 (2008ag).

DOE updated its model from TSPA Model v5.000 to v5.005, with most validation and model stability analyses performed with TSPA Model v5.000, but the annual doses reported in the SAR are based on TSPA Model v5.005 (SAR p. 2.4-76 to 78). DOE compared the effect of the change from version v5.000 to v5.005 and documented those analyses in SNL Figures 7.3.1-17[a] to 7.3.3-13[a] (2008ag). Although the stability analyses were not repeated, the comparisons indicate a similar numerical behavior of versions v5.000 to v5.005, and, thus, DOE stated that the same conclusions, with regard to stability, apply to version v5.005. DOE computed a range for the overall average annual dose using bootstrap analyses, compared the results of these analyses in SAR Figures 2.4-53 and 54, and concluded that 300 epistemic realizations were sufficient to estimate the average annual dose and that the results of TSPA Model v5.005 are statistically stable (SAR Section 2.4.2.2.2.2, p. 2.4-82).

2.2.1.4.1.3.4.2 NRC Staff Evaluation of Statistical Stability of Average Annual Dose

The NRC staff reviewed DOE's TSPA model and analytic results, as well as the SAR, and notes the calculated overall average annual dose curve is statistically stable because

- The overall average annual dose curve is reasonably stable with respect to the different approaches for representing epistemic and aleatory uncertainties (i.e., the average annual dose does not significantly change under the different approaches; for example see SAR Figures 2.4-38, 2.4-37 to 52, 2.4-55 to 61)

- Model updates from TSPA Model v5.000 to v5.005 caused only a moderate change in the magnitude of the overall average annual dose; the same results with respect to average annual dose stability are expected to apply to both versions v5.000 and v5.005 {i.e., the model updates do not cause different numerical model behavior in regard to statistical stability; for example, see SNL Figures 7.3.1-17[a] to 7.3.3-13[a] (2008ag)}

Finally, the NRC staff's confirmatory calculation provided further confidence that DOE's TSPA results were consistent with the model abstractions and capabilities of the barriers important to waste isolation described in the SAR. The NRC staff reviewed and noted the model abstractions, including uncertainties; scenario probabilities; the technical basis for excluding features, events, and processes (FEPs); and the description of the capabilities of the barriers important to waste isolation (see TER Sections 2.2.1.1–2.2.1.3) are reasonable. The NRC staff's confirmatory calculations identified that only a limited number of performance attributes (e.g., failure rate of the waste package, seepage flux into the waste package, solubility limits, and retardation in the saturated and unsaturated zones) had the potential to significantly alter the resulting average annual dose. As described in TER Section 2.2.1.4.3.3.1.2, the effect of uncertainties on DOE's average annual dose curve is limited. As addressed previously, DOE incorporated the model uncertainties into the analyses and the analyses were shown to converge to a stable solution. The NRC staff determined the analytic models reviewed in the previous 16 TER chapters are reasonable in that they were technically sound and provide a suitable representation of the performance of repository performance (i.e., the radiological consequences for the Yucca Mountain facility would not be significantly underestimated).

2.2.1.4.1.4 NRC Staff Conclusions

The NRC staff notes that DOE's representation of repository performance in its Total System Performance Assessment (TSPA) for the individual protection calculation is consistent with the guidance in the YMRP. The NRC staff also notes that the DOE technical approach for its TSPA and the TSPA results, including the average annual dose values, discussed in this chapter are reasonable.

2.2.1.4.1.5 References

BSC. 2005ak. "Saturated Zone Flow and Transport Model Abstraction." MDL–NBS–HS–000021. Rev. 03. AD 01, AD 02, ERD 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

CNWRA and NRC. 2008aa. "Risk Insights Derived From Analyses of Model Updates in the Total-system Performance Assessment Version 5.1 Code." ML082240343. San Antonio, Texas: CNWRA.

CRWMS M&O. 1998aa. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada." WBS 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O. 1996aa. "Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada." BA0000000–01717–2200–00082. Rev. 0. Las Vegas, Nevada: CRWMS M&O.

DOE. 2010ai. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.3.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.5, Set 1 and (Safety Analysis Report Sections 2.3.2 and 2.3.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 1." Letter (February 2) J.R. Williams to J.H. Sulima (NRC). ML100340034. Washington, DC: DOE, Office of Technical Management.

DOE. 2009an. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.1), Safety Evaluation Report Vol. 3, Chapter 2.2.1.1, Set 1." Letter (February 6) J.R. Williams to J.H. Sulima (NRC). ML090400455. Washington, DC: DOE, Office of Technical Management.

DOE. 2009bj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.4) Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML092110472. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cb. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.2, Table 2.2-5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.2.1, Set 5." Letter (June 5) J.R. Williams to J.H. Sulima (NRC). ML091590581. Washington, DC: DOE, Office of Technical Management.

DOE. 2009ct. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Sections 2.3.2, 2.3.3, and 2.3.5), Safety Evaluation Report Vol. 3, Chapter 2.2.1.3.6, Set 2." Letter (July 20) J.R. Williams to J.H. Sulima (NRC). ML0920204130 and ML092020410. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. ML051360241. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC and CNWRA. 2011aa. "Documentation of Analyses in Support of the NRC Staff's Technical Review of DOE's Total System Performance Assessment (TSPA) Calculations." ML110831032. Washington, DC: NRC.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. Las Vegas, Nevada: Sandia National Laboratories.

CHAPTER 18

2.2.1.4.2 Human Intrusion Calculation

2.2.1.4.2.1 Introduction

This section of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) Total System Performance Assessment (TSPA) for the human intrusion calculation, as presented in DOE's Safety Analysis Report (SAR) Section 2.4.3 (DOE, 2008ab). The geologic record provides a basis for evaluating the likelihood of geologic processes and events, but there is no similar record of extended duration that can be used to constrain either the probability that human intrusion could occur or the characteristics of such intrusion. Regulations specify that the potential effects of human intrusion on waste isolation be considered when evaluating repository performance. The NRC staff's review evaluates repository performance if its barriers are breached by a human intrusion.

2.2.1.4.2.2 Evaluation Criteria

Requirements for the human intrusion calculation are specified at 10 CFR 63.321(a) and 63.322. Other aspects of the regulations applicable to the performance assessments for the individual protection, human intrusion, and separate groundwater protection calculations are described in TER Section 2.2.1.4.1.2 (Evaluation Criteria) under the individual protection calculation. Accordingly, the human intrusion calculation considers when a human intrusion might occur and the consequences of the human intrusion. In particular, the human intrusion calculation

- Considers the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without the drillers recognizing it [10 CFR 63.321(a)]

- Assumes (i) there is a single human intrusion as a result of exploratory drilling for groundwater, (ii) the intruders drill a borehole directly through a degraded waste package into the uppermost aquifer underlying the Yucca Mountain repository, (iii) the drillers use the common techniques and practices that are currently employed in exploratory drilling for groundwater in the region surrounding Yucca Mountain, (iv) careful sealing of the borehole does not occur—instead, natural degradation processes gradually modify the borehole, (v) no particulate waste material falls into the borehole; (vi) the exposure scenario includes only those radionuclides transported to the saturated zone by water (e.g., water enters the waste package, releases radionuclides, and transports radionuclides by way of the borehole to the saturated zone), and (vii) no releases are included which are caused by unlikely natural processes [10 CFR 63.322]

- Calculates the dose that the reasonably maximally exposed individual (RMEI) receives, as a result of the human intrusion [10 CFR 63.321(b)]

The NRC staff's review followed the guidance in the Yucca Mountain Review Plan (YMRP) Section 2.2.1.4.2c (NRC, 2003aa) for human intrusion. The relevant acceptance criteria in the YMRP address the timing of an intrusion event, the representation of the human intrusion event in the Total System Performance Assessment, and the approach for estimating the annual dose to the reasonably maximally exposed individual.

In addition, the NRC staff reviewed DOE's description of the human intrusion event as part of its review of events that were included in the performance assessment in TER Section 2.2.1.2.2.3.

2.2.1.4.2.3 Technical Evaluation

The performance assessment used to estimate the dose for the human intrusion calculation is subject to specific criteria regarding the determination of the timing of the human intrusion and assumptions with respect to the nature and extent of the intrusion scenario. Accordingly, the performance assessment for the human intrusion calculation is somewhat different than the performance assessment used for the individual protection calculation. The two performance assessments are expected to differ because the performance assessment used to evaluate the human intrusion scenario includes disruption of the repository due to a postulated human intrusion event. However, those portions of the performance assessment not affected by the specifications for the human intrusion scenario would be expected to be the same as the performance assessment used for individual protection (e.g., transport of radionuclides in the saturated alluvium, characteristics of the biosphere are not affected by the postulated human intrusion event).

The NRC staff review includes a determination that the Total System Performance Assessment for human intrusion is identical to the Total System Performance Assessment for individual protection, except that it assumes the occurrence of the postulated human intrusion scenario. As a result, the NRC staff review of DOE's performance assessment for the human intrusion scenario evaluates (i) whether or not the performance assessment used for the human intrusion scenario is the same as the performance assessment used for individual protection (i.e., except for the representation of the human intrusion scenario, there are no differences between the performance assessment used for individual protection and the performance assessment used for human intrusion that would result in a significant underestimation of the peak dose for the human intrusion scenario) and (ii) whether or not those portions of the performance assessment for the human intrusion scenario are adequately represented in the performance assessment. Those portions of the Total System Performance Assessment for human intrusion that are identical to the Total System Performance Assessment for individual protection (e.g., biosphere) are evaluated as part of the review of the Total System Performance Assessment for the individual protection calculation (TER Section 2.2.1.4.1) and are not duplicated in this section.

The NRC staff's evaluation involves reviewing DOE's Safety Analysis Report (SAR), Total System Performance Assessment (TSPA) Analysis Model Report, and the TSPA model files including intermediate results.

The NRC staff review entails determining whether

- DOE's selection of the earliest time for the human intrusion to occur is adequately supported (TER Section 2.2.1.4.2.3.1)

- The performance assessment for the human intrusion calculation provides a credible representation of the human intrusion scenario (TER Section 2.2.1.4.2.3.2)

- Dose limits are statistically stable [e.g., increasing the number of simulations (statistical sample size) performed with DOE's TSPA model is not expected to significantly change the calculated average dose] (TER Section 2.2.1.4.2.3.3)

2.2.1.4.2.3.1 Timing of Human Intrusion Event

Description of DOE Approach

DOE made a determination of the earliest time at which a driller would penetrate a waste package without recognition (e.g., that a metal object had been contacted rather than rock), which is referred to as the human intrusion event. In SAR Section 2.4.3.2, DOE identified general corrosion as the process that, given sufficient time, could cause significant degradation of the drip shield and waste package such that drilling performance would most likely not be affected by the presence of the drip shield and waste package. DOE determined that there is only a 0.0001 percent chance that the drip shield will fail by corrosion before approximately 230,000 years under nominal conditions. DOE also determined that the waste package has only a 5 percent chance of failure (i.e., significant degradation or thinning of the walls of the waste package) from general corrosion prior to 600,000 years. On the basis of these results, DOE selected 200,000 years as the earliest time the waste package would degrade sufficiently that a human intrusion could occur without the drillers recognizing it. DOE considered this a conservative approach because the waste package is estimated to have experienced limited degradation due to corrosion (i.e., waste package to be substantially intact) by that time.

DOE also evaluated other events that might affect the timing of the human intrusion event. DOE did not consider unlikely natural processes and events (i.e., those events with less than 1 chance in 100,000 per year of occurring) in the evaluation of human intrusion. DOE evaluated the likelihood of early undetected defects, igneous events, and seismic events. For early undetected defects and igneous disruptive events, DOE determined that the likelihood was less than the limit for likely events. For seismic events, DOE determined that damage to either the drip shield or waste package that might compromise the structural integrity of drip shield or waste package (e.g., rupture or framework buckling of the drip shield, punctures and ruptures of the waste package) is also less than the limit for likely events (SAR Section 2.4.3.2.2, p. 2.4-303 and 304). For seismic damage that is considered likely to occur [i.e., stress corrosion cracking (SCC) of the drip shield or waste package], DOE asserted such damage would not be sufficient to prevent the driller from recognizing it (SAR Section 2.4.3.2.2, p. 2.4-303 and 304). In summary, DOE concluded that events such as early failures and igneous and seismic events are unlikely or would not cause enough damage to the drip shield and waste package to prevent the driller from recognizing the damage.

NRC Staff Evaluation of Timing of Human Intrusion Event

The NRC staff evaluated DOE's technical basis supporting the selection of the time of occurrence of human intrusion and notes that it is reasonable for the following reasons.

- General corrosion, which uniformly thins the entire surface of a material, represents a degradation process that could eventually thin or reduce the thickness of the outer barrier to the extent that a driller may not recognize the presence of a waste package. DOE has shown that (i) after 1 million years, approximately 10 percent of the waste packages, on average, are failed due to general corrosion of the waste package and (ii) prior to 400,000 years, less than 0.01 percent of commercial spent nuclear fuel waste packages and approximately 0.1 percent of codisposal waste packages are breached, as described in DOE Enclosure 1, Figures 9 and 10 (2009bj). Additionally, after 1 million years, approximately 1 percent of the surface area of the waste package is breached. The NRC staff's review of the DOE model for general corrosion of the waste package outer barrier that was implemented in the Total System Performance Assessment noted that the DOE model for general corrosion of the waste package outer barrier is reasonable and that DOE provided technical support for its calculations of the timing and magnitude of waste package breach by general corrosion (see TER Section 2.2.1.3.1.3.2.1).

- Stress corrosion cracking generally refers to a process whereby cracks form in metals or alloys in a corrosive environment and under sustained tensile stresses. Using the TSPA code, DOE calculated that even if there is sufficient stress to initiate and propagate cracks, the breached area of the waste package will be limited by the small crack size and density. The seismic ground motion modeling case, which included unlikely seismic events, resulted in less than 0.1 percent of the surface area damaged due to SCC over the 1-million-year period, as identified in DOE Enclosure 1, Figures 3, 4, 7, and 8 (2009bj). The NRC staff's review of the DOE models for stress corrosion cracking of the waste package outer barrier that were implemented in the TSPA code noted that DOE accounts for stress corrosion cracking of the waste package outer barrier in the TSPA code (see TER Section 2.2.1.3.1.3.2.3). Given the limited surface area of the waste package affected by stress corrosion cracking of the waste package, the NRC staff notes that the assumption that such degradation would not prevent the driller from recognizing the presence of the waste package is reasonable (i.e., general corrosion of the waste package rather than stress corrosion cracking could prevent the driller from doing so).

- Although the igneous intrusion modeling case assumes that the waste package and drip shield are failed when the event occurs and it might be assumed that the extent of the damage to the waste package would prevent the driller from recognizing it, the probability for the igneous intrusion is considered an unlikely event (i.e., those that are estimated to have less than 1 chance in 100,000 per year of occurring and at least 1 chance in 100 million per year of occurring) and, therefore, is excluded from the analysis of human intrusion. The NRC staff noted that (i) the preponderance of information indicates that the mean annual probability for igneous disruption of the proposed repository by a basaltic dike (intrusive case) is on the order of 1 in 100 million per year to 1 in 10 million per year and (ii) mean probability values significantly higher (i.e., 1 in 1 million per year) or lower (i.e., 1 in 1 billion per year) than this range are not consistent with past patterns of activity in the Yucca Mountain region and therefore are not considered credible (see TER Section 2.2.1.2.2.3.1).

- Seismic damage to the waste package due to tensile tearing (i.e., rupture and puncture of the waste package that represent relatively large openings in the waste package) could damage significantly more surface area of the waste package than seismically induced stress corrosion cracking, which represents relatively small openings in the waste package (see TER Sections 2.2.1.4.1.3.3.1.1.2 and 2.2.1.4.1.3.3.1.1.3). However, DOE estimated that the probability of seismic events of sufficient magnitude to cause such damage is less than 1 chance in 100,000 per year and therefore designated seismic damage as an unlikely event that can be excluded from the human intrusion scenario. The NRC staff noted that DOE's exclusion of waste package rupture was appropriately supported by kinematic analyses that considered the mechanical properties of the waste package, impact velocities of the waste package during seismic events, and degradation of drifts and drip shields (see TER Section 2.2.1.3.2.7).

Early failure of the drip shield and waste package could be a factor affecting the timing of an intrusion event. DOE estimated a probability for early failure of a drip shield that has less than 1 chance in 100,000 and therefore designated it as an unlikely event that can be excluded from the human intrusion scenario. The NRC staff noted that the probability for drip shield failure (i.e., mean value of approximately 2 chances in 1 million) based on the methodology used to estimate the probability of damage to the drip shield and data from industrial analogues for fabrication and handling of the drip shield is reasonable (see TER Section 2.2.1.2.2.4). DOE assumed the time of the human intrusion event occurs after the drip shield fails and did not take credit for any delay in the time for the human intrusion event due to the waste package being intact. This approach would not underestimate the dose. The NRC staff noted that this approach for consideration of early waste package failure is reasonable because the human intrusion is assumed to penetrate the waste package at the same time the drip shield fails.

2.2.1.4.2.3.2 Representation of Intrusion Event

Description of DOE Approach

DOE developed a separate performance assessment to evaluate the consequences of a postulated human intrusion event assumed to occur 200,000 years after permanent closure. The key elements of the postulated human intrusion event are the effects of the borehole on seepage into the waste package, release of radionuclides from the waste package, and transport of radionuclides through the unsaturated zone to the saturated zone.

The performance assessment for individual protection is used for the human intrusion analysis, including the identical sampling approach for treating uncertainty (i.e., Latin hypercube sampling). DOE modified its performance assessment for individual protection to represent human intrusion. Specifically, the performance assessment for human intrusion

- Does not include unlikely events (e.g., igneous activity or faulting)

- Assumes that damage to a single waste package occurs at 200,000 years and is the result of drill bit penetration with a cross-sectional area of 0.0324 m^2 [0.349 ft^2]; the area is based on the cross section of a borehole with a diameter of 20.3 cm [8 in]

- Assumes that seepage water enters the waste package through the borehole

- Assumes the borehole is degraded and filled with rock debris

- Assumes that releases from the waste package are passed into a fracture pathway that is assumed to exist in the borehole all the way to the saturated zone (SAR p. 2.4-296)

- Assumes radionuclides move with the flowing water down the borehole fracture to the saturated zone and are slowed only due to matrix diffusion of dissolved radionuclides from the fracture into the rock matrix

- Considers only radionuclides transported by water from the waste package to the saturated zone by way of the borehole in the exposure scenario

The quantity of water that enters the waste package and matrix diffusion in the borehole are key aspects of the representation of the human intrusion event that affect the estimated doses (e.g., infiltration was identified as an important parameter affecting the expected dose in SAR Figure 2.4-173). The quantity of water that enters the waste package through the borehole affects the release of radionuclides that are solubility limited (e.g., the release of solubility-limited radionuclides such as Np-237 will commonly be proportional to the amount of water leaving the waste package).

DOE described how the amount of water that enters the waste package through the borehole is limited to the seepage entering the borehole (deep percolation is assumed to pass directly into the borehole opening). Other processes (e.g., drift seepage water splashing on the waste package surface and entering the waste package through the hole created by the borehole) were evaluated and determined to not significantly add to the quantity of water entering the borehole, as described in DOE Enclosure 4, Section 1.1 (2009bj). DOE also described the basis for the process of matrix diffusion in the borehole, which can potentially delay both sorbing and nonsorbing radionuclide transport by providing a means for radionuclides to move from the relatively fast-flowing water in the borehole fracture into the slower moving water in the porous matrix of the rubble in the borehole. Although water in the borehole is estimated to take approximately 3 years to move through the unsaturated zone to the saturated zone, nonsorbing (I-129) and sorbing (Np-237) radionuclides are estimated to be delayed approximately 1,250 and 64,000 years, respectively, as outlined in DOE Enclosure 5, p. 8 (2009bj). DOE described in DOE p. 2 (2009cp) that this effect is due to the large effective surface area for communication between the fracture and the matrix in the degraded borehole and along the borehole.

DOE also evaluated the potential effect on repository performance if the borehole penetrated a perched water zone (i.e., groundwater separated from an underlying body of groundwater by an unsaturated zone) below the repository. If radionuclides were present in a perched water zone, the borehole penetration of a perched water zone could potentially affect the transport of radionuclides from within the perched zone to the saturated zone. DOE described that the effect of the borehole would be limited because (i) perched water zones below the repository are isolated and have limited volume, as described in DOE Enclosure 6, p. 2 (2009bj); (ii) the significance of an equivalent 20.3-cm [8-in]-diameter borehole to capture and divert any lateral flow associated with the perched water is expected to be small because the area associated with fractures in the rock is more than 10,000 times greater than the area associated with the borehole, as identified in DOE Enclosure 6, p. 5 (2009bj); and (iii) the performance assessment already includes fast transport times in fault zones, which would not be significantly influenced by another fast pathway—namely, the borehole—as outlined in DOE Enclosure 6, p. 6 (2009bj).

NRC Staff Evaluation of Representation of Intrusion Event

The NRC staff has evaluated DOE's technical basis supporting its separate performance assessment for the postulated human intrusion event. The NRC staff notes that DOE's separate performance assessment for the human intrusion event is reasonable for the following reasons.

- **Assumptions for the method of transport from the waste package are reasonable.**

  DOE's approach described on SAR p. 2.4-295 provided for radionuclide release from the waste package directly following the assumed time of the human intrusion event. The releases from the waste package pass directly into the borehole and travel down the borehole via water in an assumed continuous fracture all the way to the saturated zone {i.e., via the flow of water in a single small fracture with an approximate fracture aperture of 3 mm [0.12 in]}, which was estimated to have an average water travel time through the unsaturated zone to the saturated zone of a few years, as described in DOE Enclosure 5, p. 6 (2009bj). DOE's approach for the transport of radionuclides from a breached waste package is reasonable because the assumed continuous fracture path provides (i) a consistent method of transport among the models of the Total System Performance Assessment that connects the releases from the waste package to the saturated zone and (ii) water travel times in this continuous fracture path on the order of a few years are considered to be a conservative estimate for travel from the waste package through the unsaturated zone to the saturated zone compared to the 100-year travel times DOE has presented for fault zones, which are also characterized as a continuous "fracturelike" path through the unsaturated zone to the saturated zone [DOE Enclosure 6, p. 6 (2009bj)]. Further, travel times on the order of years represent no significant delay for water down the borehole; thus, this value would not result in the dose being underestimated.

Additionally, DOE determined the amount of seepage water that enters the waste package from the borehole, which provides the water flux for advective transport of radionuclides out of the waste package using the same deep percolation values from the performance assessment for individual protection and assuming that this seepage enters the cross-sectional area of the borehole, as outlined in SAR p. 2.4-317. This approach, which assumes all water flowing downward in the borehole enters the waste package, is reasonable because it is consistent with the movement of deep percolation vertically downward toward the repository horizon. Additionally, as DOE evaluated in DOE Enclosure 4, pp. 2–3 (2009bj), the potential for other adjacent seepage water entering the drift to enter the waste package is limited due, in part, to the limited distance that such seepage water "splashing" on the corroded waste package surface could travel and enter the borehole opening into the waste package.

- **Physical processes associated with the postulated human intrusion have been verified.**

  Matrix diffusion in the borehole is the primary means for delay of radionuclides transported from the waste package through the unsaturated zone to the saturated zone. In DOE Enclosure 5, pp. 8–11 (2009bj), DOE provided support to verify its approach for matrix diffusion within the borehole by providing a comparison with an analytical solution, which had essentially an identical match between the human intrusion approach in the performance assessment and the analytical solution. DOE also explained differences between matrix diffusion within the borehole and its approach for representing matrix diffusion within the unsaturated zone in the performance assessment for individual protection using the Active Fracture Model. DOE explained in DOE Enclosure 1, p. 1 (2009cp) that the Active Fracture Model was developed for fracture networks rather than a single fracture (as in the borehole) and, therefore, is not appropriate for the human intrusion borehole pathway. DOE's approach of representing the borehole as a single fracture is reasonable because of the limited diameter of the borehole and, as described under the previous bullet of this evaluation, the water travel time in the single fracture is on the order of a few years, representing no significant delay for water moving down the borehole.

- **The uncertainty in the results is consistent with the postulated intrusion event.**

  The results of the treatment of uncertainty, as displayed in the spread of dose curves in SAR Figures 2.4-11 and 2.4-159, are consistent with (i) the radionuclide inventory of the intruded waste package and (ii) the release and transport characteristics of the soluble, nonsorbing radionuclides (e.g., Tc-99 and I-129 are the main contributors to the peak dose shortly after the human intrusion) and less soluble, sorbing radionuclides (i.e., Pu-242 and Np-237, which are main contributors to dose long after the human intrusion occurs).

  Additionally, the NRC staff performed a confirmatory calculation to understand the magnitude of releases to the location of the RMEI due to the human intrusion event. Using the average dose curve and the average biosphere dose conversion factors from DOE's performance assessment, the NRC staff has calculated the magnitude of the releases for radionuclides most relevant to the dose calculation. For those radionuclides that do not sorb onto rock surfaces or corrosion products, especially Tc-99 and I-129, a very large fraction of the inventory for these radionuclides (on the order of 1 percent over a 100-year period) must be released to the RMEI location to produce the peak dose. Conversely, releases of radionuclides that do sorb onto rock surfaces, especially Pu-242 and Np-237, release a much smaller fraction than 0.01 percent over a 100-year period to sustain the peak dose for these radionuclides. The results of this confirmatory calculation are consistent with the postulated human intrusion event and the relevant aspects of the performance assessment, including uncertainties, DOE used in the individual protection calculation (see TER Section 2.2.1.4.1 for further details on the performance assessment used for individual protection).

- **The sampling method ensures the range for uncertain parameters are sampled.**

  DOE used the stratified Monte Carlo technique (Latin hypercube sampling) for sampling uncertain parameters, which also was used for the performance assessment for individual protection. DOE's approach to sampling ensures the sampled parameters were sampled across their range of uncertainty because (i) stratified Monte Carlo sampling is a common sampling approach used in analyses involving uncertain parameters such as waste disposal, (ii) DOE considered alternative sampling combinations (called "replicates" in SAR Section 2.4.3.3.3) that all resulted in nearly identical dose curves, and (iii) scatter plots (SAR Figures 2.4-174 and 2.4-176) showed that DOE's sampling approach produced sampled values with a range of uncertainty [the scatter plots presented sampled values for an uncertain parameter (x-axis) versus the resulting dose for the TSPA simulation that used the specific value for the parameter (y-axis)].

  DOE also evaluated the potential effects on performance from a borehole penetrating a perched water zone. The NRC staff evaluated DOE-provided information to understand how a borehole might affect the transport of radionuclides from a perched water zone. DOE explained that the potential for a borehole penetrating a perched water zone to significantly impact repository performance is limited; this is reasonable because (i) a single, 20.3-cm [8-in]-diameter borehole (cross-sectional area less than 1/10 of a square meter or approximately 1/3 square foot) filled with rubble does not represent a significant feature that could divert significant water flow relative to the unsaturated flow already occurring in faults and fractures beneath the repository footprint of approximately 5.7 million m^2 [61 million ft^2] (SAR p. 2.3.1-85) [e.g., DOE estimated in

DOE Enclosure 6, p. 5 (2009bj) that 30 percent of the total water flux below the repository reaches the water table via faults]; (ii) given the borehole would primarily affect water flow in the matrix (e.g., water flow in fractures and faults would not be that dissimilar from the continuous fracture path assumed for the borehole), the impact is limited due to the small amount of water that reaches the water table below Yucca Mountain via matrix flow [i.e., less than 20 percent of the flux at the water table is from matrix flow with the remaining flux coming from fractures and faults, as identified in DOE Enclosure 6, Table 1 (2009bj)]; and (iii) perched water is of somewhat limited areal extent {i.e., DOE estimated an equivalent area of 900 m^2 [9,700 ft^2] for the radius of a perched zone, as described in DOE Enclosure 6, p. 4 (2009bj), which occurs mostly in the northern part of the repository where fracture flow is more prevalent, as outlined in DOE Enclosure 6, p. 2 (2009bj)}.

2.2.1.4.2.3.3 Annual Dose to RMEI

Description of DOE's Approach

DOE presented the dose curve for the human intrusion scenario in SAR Figure 2.4-11. The peak of the mean dose curve is approximately 0.0001 mSv/yr [0.01 mrem/yr] shortly after the time of the intrusion (i.e., 200,000 years). DOE performed tests to determine the computational stability of the average dose curve for the human intrusion calculation (SAR Section 2.4.3.3.3). The tests (i) computed three replicates to allow for different combinations of sampled values over their parameter ranges (SAR Figure 2.4-160); (ii) increased the number of aleatory samples from 30 to 90 (SAR Figure 2.4-161), and (iii) refined the timestep scheme, as shown in SNL Figures 7.3.3-10[a] and 7.3.3-11[a] (2008ag). DOE determined that expected doses were relatively unaffected (i.e., stable) by changes in values of sampled parameters, sample size, and timestepping.

NRC Staff Evaluation of Annual Dose to RMEI

The NRC staff notes that DOE's estimated peak dose of 0.0001 mSv/yr [0.01 mrem/yr] for the human intrusion calculation is reasonable for the following reasons.

- **The TSPA for the human intrusion calculation is performed separately from the TSPA for individual protection and is consistent with the YMRP and other guidance criteria.**

  DOE developed a separate dose curve (SAR Figure 2.4-11) for the human intrusion scenario using the separate TSPA model described in SAR Section 2.4.3. The TSPA for the human intrusion calculation is reasonable because DOE (i) provided the technical basis for the relevant attributes of the human intrusion scenario (i.e., water entering the waste package through the borehole and transport with the borehole), and these were different from the performance assessment for individual protection; (ii) accounted for uncertainties in the representation of the human intrusion (see TER Section 2.2.1.4.2.3.2, NRC Evaluation, Item 2); (iii) excluded unlikely features, events, and processes (see TER Section 2.2.1.4.2.3.1, NRC Evaluation, Items 3, 4, and 5); and (iv) provided a comparison with an alternative model [i.e., the analytical solution of Sudicky and Frind described in DOE Enclosure 5, Figure 2 (2009bj)] to support the approach for matrix diffusion in the borehole.

- **The TSPA model for the human intrusion calculation assumes the specified characteristics.**

  Those portions of the performance assessment that DOE used to represent the human intrusion event included that (i) there is a single exploratory borehole that intersects the waste package providing a conduit to the saturated zone, and as a result, water enters the waste package and transports radionuclides from the intersected waste package to the saturated zone (SAR pp. 2.4-293 to 2.4-298); (ii) the borehole is depicted as a 20.3-cm [8-in]-diameter borehole, which results in a cross-sectional area of 0.0324 m^2 [0.349 ft^2], based on current drilling practices, that is assumed to be filled with rubble of collapsed host rock and not carefully sealed, and particulate waste material does not fall into the borehole (SAR pp. 2.4-293 to 2.4-298); and (iii) unlikely features, events, and processes were excluded (see TER Section 2.2.1.4.2.3.1 for the review of the excluded events).

  Those portions of the performance assessment for human intrusion that have not been "stylized" to represent the human intrusion event are consistent with the performance assessment used for individual protection. The TSPA model for the human intrusion calculation and the performance assessment for individual protection are the same (i.e., any differences would not result in a significant underestimation of the peak dose for the human intrusion calculation) regarding the following relevant portions of the performance assessment used for individual protection: releases from the waste package, where releases are affected by solubility limits; degradation rates for the waste forms and sorption onto corrosion products; radionuclide transport in the saturated zone; and the representation of the characteristics of the biosphere and RMEI, which includes biosphere dose conversion factors.

- **The estimate of the mean dose is statistically stable.**

  The NRC staff reviewed SAR Section 2.4.3 as well as the relevant information in SNL Section 7.3 (2008ag). DOE provided reasonable support for the statistical stability of the expected dose because DOE considered a range of tests (i.e., different combinations of sampled values, increased aleatory sample size, and reduced timesteps), all of which resulted in dose curves that do not change the overall result that the peak dose is on the order of 0.0001 mSv/yr [0.01 mrem/yr] [see SAR Figures 2.4-160(a) and 2.4-161 and SNL Figures 7.3.3-10(a) and 7.3.3-11(a) (2008ag)].

- **The dose estimate is consistent with the overall repository performance and the assumed characteristics of the human intrusion scenario.**

  The human intrusion scenario occurs at 200,000 years and intercepts a single waste package. Thus, it is expected that any radionuclide with a radioactive half-life on the order of 20,000 years and less, without a long-lived parent radionuclide, would decay sufficiently prior to the intrusion event to significantly limit the contribution to dose. The DOE's TSPA results for the human intrusion calculation are consistent with the concept that radionuclides with long half-lives are expected to be the largest contributors to dose [i.e., significant contributors to peak dose are Tc-99 (half-life of 213,000 years), I-129 (half-life of 15.7 million years), Pu-242 (half-life of 376,300 years), Se-79 (half-life of 1.13 million years), Cs-135 (half-life of 2.3 million years), and Np-237 (half-life 2.14 million years), as shown in SAR Figure 2.4-159]. Additionally, both the human intrusion scenario (SAR Figure 2.4-159) and the waste package early failure modeling case [SAR Figure 2.4-18(b)] result in peak doses over the million-year period on the order of .0001 mSv/yr [0.01 mrem/yr], consistent with the characteristic that the human intrusion scenario considers one waste package and the waste package early failure modeling case considers approximately one failed waste package (see TER Section 2.2.1.4.1.3.2). On the basis of these reasons, the NRC staff notes that DOE's estimated dose for the human intrusion calculation is reasonable.

2.2.1.4.2.4 NRC Staff Conclusions

NRC staff notes that DOE'S representation of repository performance in its Total System Performance Assessment (TSPA) for the human intrusion calculation is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach for its TSPA and the TSPA results, including the timing of the intrusion event and average annual dose values, discussed in this chapter are reasonable.

2.2.1.4.2.5 References

DOE. 2009bj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.4) Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML092110472. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cp. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.4.2, Set 1." Letter (October 20) J.R. Williams to J.H. Sulima (NRC). ML092940188. Washington, DC: DOE, Office of Technical Management.

DOE. 2008ab. DOE/RW–0573, "Yucca Mountain Repository License Application." Rev. 0. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: NRC.

SNL. 2008ag. "Total System Performance Assessment Model/Analysis for the License Application." MDL–WIS–PA–000005. Rev. 00. AD 01, ERD 01, ERD 02, ERD 03, ERD 04. ML090790353. Las Vegas, Nevada: Sandia National Laboratories.

(Intentionally Left Blank)

CHAPTER 19

2.2.1.4.3 Separate Groundwater Protection Calculation

2.2.1.4.3.1 Introduction

This section of the Technical Evaluation Report (TER) provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE's) Total System Performance Assessment (TSPA) for the separate groundwater protection calculation, as presented in DOE's Safety Analysis Report (SAR) Section 2.4.4 (DOE, 2008ab).
10 CFR Part 63 provides separate standards to protect the groundwater resources in the vicinity of Yucca Mountain and the approach to be taken to estimate the concentration of radionuclides in groundwater. This approach (10 CFR 63.331) is similar to that used in estimating dose to the reasonably maximally exposed individual (RMEI). There are three distinct groups of radionuclides evaluated under groundwater protection: (i) radionuclides that are characterized as alpha emitters (e.g., Np-237) (this group explicitly excludes radon and uranium); (ii) radionuclides that are characterized as beta- and photon-emitting radionuclides (e.g., I-129, Tc-99); and (iii) the combined concentration of Ra-226 and Ra-228 released from the repository and the natural background levels of Ra-226 and Ra-228 in the groundwater. There are a number of similarities in the performance assessment used for the individual protection calculation and the performance assessment used for the separate groundwater protection calculation.

2.2.1.4.3.2 Evaluation Criteria

Separate groundwater protection standards for the initial 10,000 years after closure of the repository are in 10 CFR 63.331. Provisions in 10 CFR 63.331, 63.332, and 63.342 specify constraints for the performance assessment used for the groundwater protection calculation and the determination of the "representative volume" of groundwater (i.e., the volume of water used to estimate the concentration of radionuclides). Other criteria applicable to the performance assessments for the individual protection, human intrusion, and separate groundwater protection calculations are described in TER Section 2.2.1.4.1.2 (Evaluation Criteria) under the individual protection calculation. The criteria specific to the separate groundwater protection calculation are summarized next.

Performance Assessment for Groundwater Protection Calculation

- The performance assessment excludes unlikely features, events, and processes (FEPs) that are estimated to have less than 1 chance in 100,000 per year of occurring [10 CFR 63.342(b)].

- The performance assessment excludes the consideration of human intrusion (i.e., representing undisturbed performance) (10 CFR 63.331).

Representative Volume

- The representative volume is the volume of groundwater that would be withdrawn annually from an aquifer containing less than 10,000 mg of total dissolved solids per liter of water [10 CFR 63.332(a)].

- The concentration of radionuclides that will be released from the Yucca Mountain repository is determined for the representative volume of groundwater [10 CFR 63.332(a)].

- The position and dimensions of the representative volume use average hydrologic characteristics that include the highest concentration level in the plume of contamination in the accessible environment [10 CFR 63.332(a)(1) and (2)].

- The representative volume contains 3,000 acre-ft {about 3,714,450,000 L [977,486,000 gal]} of water [10 CFR 63.332(a)(3)].

Concentrations To Be Estimated for the Separate Groundwater Protection Calculation

The following concentrations are to be estimated for the separate groundwater protection calculation (10 CFR 63.331):

- The combined concentration of Ra-226 and Ra-228 from repository releases (including natural background radiation presently in groundwater at Yucca Mountain)

- The concentration for gross alpha activity (including Ra-226 but excluding radon and uranium), including natural background radiation presently in groundwater at Yucca Mountain (Np-237 is an example of an alpha-emitting radionuclide)

- The combined concentration of beta- and photon-emitting radionuclides from repository releases and the associated dose per year to the whole body or any organ {on the basis of drinking 2 L [0.53 gal] of water per day from the representative volume} (Tc-99 and I-129 are examples of beta- and photon-emitting radionuclides)

The performance assessment for the individual protection calculation is the same performance assessment used for the groundwater protection calculation (i.e., except for differences due to the regulatory requirement that unlikely events are not to be included in the performance assessment used for groundwater protection, there are no differences between the performance assessment used for estimating individual dose and the performance assessment used for estimating concentrations for the separate groundwater protection calculation that would result in a significant underestimation of the concentration of radionuclides in groundwater). As a result, the NRC staff review of DOE's groundwater protection analysis focused on DOE's determination of the representative volume and the approach for estimating the radionuclide concentrations. The performance assessment for individual protection is reviewed in TER Section 2.2.1.4.1.

The NRC staff's review followed the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa). Because DOE assumed that all radionuclides which reach the accessible environment in a given year are included in the annual water demand of 3,000 acre-ft, YMRP Section 2.2.1.4.3.1 states that the NRC staff may conduct a simplified review.

2.2.1.4.3.3 Technical Evaluation

The NRC staff review of DOE's separate groundwater protection calculation focused on those portions of the analysis that are distinct to the groundwater protection analysis. Specifically, the NRC staff review focused on DOE's approach for including the highest concentration level of the plume in the representative volume, the dimensions of the representative volume, and the concentration of radionuclides in groundwater.

2.2.1.4.3.3.1 Representative Volume Location

DOE used the same performance assessment model for evaluating the separate groundwater protection calculation as it used for evaluating the individual protection calculation in the sense that the model abstractions for flow paths in the saturated zone and radionuclide transport in the saturated zone are the same. However, DOE excluded the consideration of unlikely FEPs from the performance assessment used for groundwater protection (i.e., igneous activity and low probability seismic events are excluded). The location of the representative volume of groundwater was consistent with the approach used for the determining the pathway for radionuclide transport to the location of the reasonably maximally exposed individual, which is approximately 18 km [11 mi] south of the repository, as identified in the Safety Analysis Report (SAR) Volume 2, p. 2.1-1 (DOE, 2008ab). Additionally, DOE used the same approach for determining the concentration of radionuclides in groundwater for both the separate groundwater protection and individual protection calculations {i.e., the annual average radionuclide concentration, due to releases from the repository, was determined by assuming that all radionuclides that reach the RMEI location in a given year are included in 3,000 acre-ft, which is the annual water demand for the individual protection calculation and the representative volume for groundwater (SAR Section 2.4.4)}.

NRC Staff Evaluation of the Representative Volume Location

The NRC staff has reviewed SAR Section 2.4.4 and noted that DOE’s approach for identifying the location of the representative volume and including the highest concentration within the plume in the accessible environment is reasonable for the following reasons:

- The location of the representative volume is approximately 18 km [11 mi] south of the repository. This location is based on DOE’s specification for the postclosure controlled area, which extends the southern boundary of the controlled area to 36°40’13.6661” north latitude (SAR Volume 2, p. 2.4-7). Thus the location of the representative volume is in the accessible environment immediately outside the controlled area.

- The location of the representative volume has been determined consistent with the radionuclide transport paths in the performance assessment used for the individual protection calculation because the same radionuclide transport paths were used in the performance assessment for individual protection and the performance assessment for groundwater protection (SAR Volume 2, p. 2.4-337).

- The location of the representative volume ensures that all radionuclides released to the accessible environment are considered in the assessment because DOE’s Total System Performance Assessment (TSPA) assumes all radionuclides are captured in the representative volume (i.e., “total radionuclide capture,” SAR Volume 2, p. 2.4-337).

- The highest concentration level of radionuclides in the plume of contamination in the accessible environment is included in the representative volume because all radionuclides (i.e., the entire plume of contamination, which includes the highest concentration level of radionuclides) released into the accessible environment are included in the representative volume that is annually withdrawn (SAR Volume 2, p. 2.4-337).

2.2.1.4.3.3.2 Representative Volume Dimensions

DOE used the models and assumptions to estimate flow and transport paths in the saturated zone for the individual protection calculation and for estimating the dimensions of the representative volume. DOE estimated, using the slice of the plume method, that dimensions of a width of 3,000 m [9,842 ft], a depth of 200 m [656 ft] and a length of 30 m [98 ft] in the direction of groundwater flow would include all the simulated flow paths of radionuclides crossing into the accessible environment (SAR Volume 2, p. 2.4-337). DOE estimated these dimensions using average properties for hydrologic parameters such as groundwater flow rate and alluvium flow porosity (SAR Volume 2, p. 2.4-337). DOE calculated the representative volume with these dimensions that yielded a volume of approximately 3,000 acre-ft.

DOE also presented a more detailed depiction of the cross section of the plume in the accessible environment to further support the dimensions of the representative volume. The more detailed analysis was based on numerous particle tracks, provided in DOE Enclosure 7, Figure 1 (2009bj), representing potential release points for repository releases to the saturated zone using the saturated zone site-scale flow model. Although the cross section of the plume, based on the particle traces, is not a rectangular shape, DOE estimated that a rectangular shape of approximately 3,300 m [10,827 ft] in width (horizontally) by 220 m [722 ft] in depth (vertically) would enclose the horizontal and vertical extent of the plume cross section {an area of 726,000 m^2 [7.8 million ft^2]}. DOE estimated that approximately 40 percent of this rectangular shape did not contain any significant portion of the plume; thus, DOE estimated a cross-sectional area of 435,000 m^2 [4.7 million ft^2] given the irregularities of the shape produced by the particle traces depicted in DOE Enclosure 7, Figure 1 (2009bj). DOE's simple rectangular approximation {i.e., 3,000 by 200 m [9,842 by 656 ft]} results in a cross-sectional area of 600,000 m^2 [6.4 million ft^2], which provides a value between the two values calculated from the particle tracks {one a rectangular shape of 726,000 m^2 [7.8 million ft^2] and the other an irregular shape of 435,000 m^2 [4.7 million ft^2]}. The third dimension of the representative volume was selected to obtain the volume of 3,000 acre-ft. DOE calculated the third dimension, or the length parallel to the flow direction (i.e., perpendicular to the cross section), to be approximately 34.4 m [113 ft] on the basis of the cross-sectional area of 600,000 m^2 [6.4 million ft^2] and an average effective porosity of 0.18, as identified in DOE Enclosure 7, p. 4 (2009bj).

DOE used water quality data from the Alluvial Testing Complex (SAR Volume 2, p. 2.4-334) to determine that there were fewer than 500 mg/L [500 ppm] of total dissolved solids in the aquifer in the accessible environment.

NRC Staff Evaluation of the Representative Volume Dimensions

The dimensions of the representative volume are to include the highest concentration level in the plume of contamination. DOE determined (i) the dimensions of the representative volume using the slice of the plume approach and (ii) these dimensions are sufficient to capture all the releases into the accessible environment. The NRC staff reviewed SAR Section 2.4.4 and determined that the dimensions of the representative volume are reasonable because (i) the dimensions are sufficient to capture the entire radionuclide plume and thus include the highest concentration levels in the plume and (ii) the dimensions are supported by particle tracks that used the hydrologic characteristics of the site and releases from the engineered barrier system.

Specifically, the NRC staff notes that

- The representative volume of groundwater analyzed by DOE is within an aquifer containing fewer than 10,000 mg/L of total dissolved solids and no more than 3,000 acre-ft

- DOE estimated the dimensions of the representative volume on the basis of the slice of the plume method

- DOE used (i) average hydrologic characteristics representative of the aquifers along the flow paths in the saturated zone and (ii) the flow paths predicted by the saturated zone site-scale flow model used for the performance assessment (SAR Volume 2, p. 2.4-337)

- The representative volume of groundwater of 3,000 acre-ft is consistent with the water usage of the reasonably maximally exposed individual (i.e., annual water demand of 3,000 acre-ft)

- The dimensions of the representative volume (i) do not exclude any radionuclides from the estimate of the concentration of radionuclides in the representative volume (i.e., all radionuclides are assumed to lie within the dimensions of the representative volume that is annually withdrawn) and (ii) are reasonably consistent with the estimated shape of the contaminant plume

Additionally, DOE used a particle tracking approach to support the dimensions of the representative volume. The NRC staff evaluated DOE's particle tracking approach and notes the following:

- DOE's particle tracking approach released particles over the entire repository footprint, which provided the initial areal extent of the potential plume. This initial release area is consistent with the performance of the repository regarding the potential for damaged packages over the entire footprint. The performance assessment for the groundwater protection calculation is primarily influenced by the seismic ground motion modeling case (see SAR Figure 2.4-181). DOE presented information in the SAR that shows the seismic ground motion modeling case results in a significant number of codisposal waste packages (e.g., hundreds of waste packages) being breached due to stress corrosion cracks prior to 10,000 years (see SAR Figures 2.4-19 and 2.4-77). TER Sections 2.2.1.3.2 and 2.2.1.4.1 further detail the extent of damage to codisposal waste packages in the seismic ground motion modeling case.

- DOE's particle tracking approach used the saturated zone site-scale model consistent with the performance assessment abstraction for flow paths in the saturated zone (see TER Section 2.2.1.3.8 regarding NRC staff review of the saturated zone site-scale flow model). Thus, the spreading of the plume during transport to the accessible environment uses the same hydrologic characteristics reviewed for the performance assessment used for individual protection (see TER Sections 2.2.1.3.8 and 2.2.1.3.9).

- The two values DOE estimated from the detailed analysis for the cross-sectional area of the representative volume {one a rectangular shape of 726,000 m^2 [7.8 million ft^2] and the other an irregular shape of 435,000 m^2 [4.7 million ft^2]} bound the value of 600,000 m^2 [6.4 million ft^2] DOE specified for the representative volume. Given dimensions of this magnitude {i.e., hundreds of thousands of square meters [millions of square feet]}, it is reasonable to assume a significant portion of the releases of radionuclides into the accessible environment would be captured in the representative volume. DOE assumed all the radionuclides are released into the representative volume; thus the concentration of radionuclides in the representative volume did not change based on changes to the dimensions of the representative volume for the range of values estimated from the detailed analysis.

2.2.1.4.3.3.3 Concentration of Radionuclides in the Representative Volume

DOE determined the average concentration of radionuclides, due to repository releases, by assuming the annual releases of radionuclides were all included in the representative volume of 3,000 acre-ft and determined the dose to the whole body and individual organs for the beta- and photon-emitting radionuclides on the basis of drinking 2 L [0.53 gal] per day of water at the concentration level estimated for the representative volume (SAR Section 2.4.4.1.1.4). DOE also estimated the natural background level of radioactivity presently in the groundwater at Yucca Mountain for Ra-226, Ra-228, and the alpha-emitting radionuclides, excluding radon and uranium (SAR Section 2.4.4.1.1.3).

DOE estimated the combined concentrations for Ra-226 and Ra-228, due to releases from the repository and the natural background radiation presently in the groundwater at Yucca Mountain, was 0.5 pCi/L with the largest contribution coming from natural background radiation (i.e., the largest annual release of Ra-226 and Ra-228 into the representative volume from the repository was estimated to be almost 1 million times less than the natural background level).

DOE estimated the concentration for the gross alpha activity, due to releases from the repository and the natural background radiation presently in the groundwater at Yucca Mountain (excluding radon and uranium), was 0.5 pCi/L with the largest contribution coming from natural background radiation (i.e., the largest annual release of the relevant alpha-emitting radionuclides into the representative volume from the repository was estimated to be more than 1,000 times less than the natural background levels).

DOE estimated the dose from beta- and photon-emitting radionuclides, due to releases from the repository, to be 0.0006 mSv/yr [0.06 mrem/yr] for the whole body and the largest dose to any organ to be 0.0026 mSv/yr [0.26 mrem/yr] (e.g., dose to the thyroid from I-129) as result of drinking 2 L [0.53 gal] of water per day assumed to be at a concentration level of radionuclides in the representative volume. (Natural background radiation is not considered for beta- and photon-emitting radionuclides in the separate groundwater protection calculation.)

NRC Staff Evaluation of the Concentration of Radionuclides in the Representative Volume

The NRC staff has reviewed SAR Section 2.4.4 and determines that DOE's analysis for the level of radioactivity in the representative volume is reasonable. The NRC staff notes that

- The same performance assessment used for the individual protection calculation was used for the separate groundwater protection calculation (see TER Section 2.2.1.4.1 for details regarding the review of the performance assessment used for individual protection)

- Unlikely natural FEPs are excluded from the performance assessment for the separate groundwater protection calculation (SAR Volume 2, p. 2.4-328)

- The effects of human intrusion are not included in the performance assessment for the separate groundwater protection calculation (i.e., undisturbed performance was evaluated) (SAR Volume 2, p. 2.4-329)

- The average concentrations from repository releases are consistent with the performance assessment used for the individual protection calculation for the initial 10,000 years (e.g., number and types of waste package failures) and the specific constraints on the performance assessment used for the separate groundwater protection calculation (e.g., exclusion of unlikely FEPs) (see TER Section 2.2.1.4.1 for further details)

- The average concentrations from repository releases are determined by dividing the annual flux of radionuclides crossing into the accessible environment by the representative volume of 3,000 acre-ft that is withdrawn annually (SAR Volume 2, p. 2.4-329)

- DOE estimated the mean natural background activity concentration for the combined Ra-226 and Ra 228 and for the relevant alpha-emitting radionuclides (i.e., excluding radon and uranium) using samples collected in the vicinity of the RMEI location (SAR Section 2.4.4.1.1.3) and other locations

- Dose estimates for beta- and photon-emitting radionuclides consider the highest dose among the whole body or any organ on the basis of drinking 2 L [0.53 gal] per day from the representative volume

2.2.1.4.3.4 NRC Staff Conclusions

NRC staff notes that DOE's representation of repository performance in its Total System Performance Assessment (TSPA) for the separate groundwater protection calculation is consistent with the guidance in the YMRP. NRC staff also notes that the DOE technical approach for its TSPA and the TSPA results, including the concentrations of radionuclides and the location and dimensions of the representative volume, discussed in this chapter are reasonable.

2.2.1.4.3.5 References

DOE. 2009bj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.4) Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). Enclosures (8). ML092110472. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2008ab. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 00. ML081560400. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: NRC.

CHAPTER 20

2.5.4 Expert Elicitation

2.5.4.1 Introduction

This chapter of the Technical Evaluation Report (TER) evaluates the information provided in the U.S. Department of Energy (DOE) Safety Analysis Report (SAR) for uses of expert elicitation. DOE's uses are described in SAR Section 5.4 (DOE, 2009av).

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts. The U.S. Nuclear Regulatory Commission (NRC) routinely accepts, for review, expert judgments used to evaluate and interpret the factual bases of SARs. In previous interactions between NRC and DOE, NRC staff acknowledged that DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, DOE used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information such as data collection, analyses, and experimentation. In this context, the NRC staff reviewed DOE's use of expert elicitation regarding the proposed geologic repository at Yucca Mountain.

In supporting its SAR, DOE presented the results of three expert elicitations in the areas of seismic hazard (SAR Section 2.2.2.1), igneous activity (SAR Sections 1.1.6.2, 2.2.2.2, and 2.3.11), and saturated zone flow and transport (SAR Section 2.3.9.2). SAR Section 5.4 summarized DOE's bases for its assertion that these elicitations were conducted in a manner that is generally consistent with NRC guidance. In conducting its review of DOE's use of expert elicitation, NRC staff sought to verify that DOE followed the process suggested in NUREG–1563 (NRC, 1996aa), or some other equivalent stepwise process, such as that outlined in NUREG/CR–6372 (NRC, 1997aa).

2.5.4.2 Evaluation Criteria

10 CFR 63.21(c)(19) requires that the SAR include an explanation of how expert elicitation was used. In 1996, the NRC staff published guidance for the use of expert elicitation in NUREG–1563 (NRC, 1996aa). NUREG–1563 provided general guidelines for deciding whether a formal expert elicitation would be useful, and suggested a nine-step procedure that could serve as one acceptable process to conduct an elicitation. The guidance explicitly states that the suggested procedure was not provided with the intent that it be rigidly applied. Rather, the guidance in NUREG–1563, p. 22 (NRC, 1996aa) provides that the suggested procedure "…should be viewed as a general framework for a formal elicitation that would be acceptable to the NRC staff."

Subsequent to the release of NUREG–1563, NRC staff published NUREG/CR–6372 (NRC, 1997aa). This document, referred to informally as the Senior Seismic Hazard Analysis Committee (SSHAC) report, or the SSHAC guidelines, provided a process for obtaining, communicating, and quantifying the uncertainties associated with elicitation received from seismic experts in the course of conducting Probabilistic Seismic Hazard Assessments (PSHAs) for commercial nuclear power plants and other critical facilities. The stepwise processes for eliciting experts described in the SSHAC guidelines for the most formal (Level 4) analysis and that which is recommended in NUREG–1563 are very similar. While presented in a slightly different order and structure (in seven steps as opposed to nine, respectively), the two documents recommend essentially the same approach for formally eliciting and documenting expert opinion. For example, the important content identified in NUREG–1563 as Step 4, "Assembly and Dissemination of Basic Information"; Step 6, "Elicitation of Judgments"; and Step 7, "Post-Elicitation Feedback" is not treated as discrete steps in the SSHAC guidelines. Instead, the SSHAC guidelines encompass the substance of all three in a single Step 5, referred to as "Group Interaction and Individual Elicitation."

NRC staff's review of DOE's use of expert elicitation was guided by the Yucca Mountain Review Plan (YMRP) Section 2.5.4 (NRC, 2003aa). YMRP Section 2.5.4.3 identifies two acceptance criteria: that DOE use NUREG–1563 or equivalent procedures and that any updated elicitations follow appropriate methods and are adequately documented. These are the only two acceptance criteria applicable to NRC staff's review of DOE's use of expert elicitation. NRC staff evaluated the techniques DOE used to conduct three expert elicitations to verify whether the elicitations either followed procedures suggested by NRC staff guidance or used equivalent procedures. DOE updated only one of the three elicitations. NRC staff evaluated the methods DOE used to update that elicitation to verify whether it was updated appropriately and adequately documented.

2.5.4.3 Technical Evaluation

This section briefly summarizes the information provided in SAR Section 5.4 for each of the three expert elicitations DOE used. The discussion at the end of this section provides NRC staff's evaluation on the basis of the guidance in the YMRP.

Probabilistic Volcanic Hazard Assessment (PVHA) Expert Elicitation

SAR Section 2.2.2.2 described the DOE approach to developing a volcanic hazard assessment for Yucca Mountain. This overall approach included an expert elicitation to develop a PVHA for Yucca Mountain. DOE conducted the expert elicitation in 1995 and published the final report in 1996 (CRWMS M&O, 1996aa). SAR Section 5.4.1 summarized DOE's bases for how its PVHA was conducted in a manner generally consistent with the nine-step procedure suggested in NUREG–1563 (NRC, 1996aa).

For PVHA, DOE empanelled 10 subject matter experts to assess the relevant technical issues, including a range of conceptual and probability models, associated uncertainties in model parameters, and model sensitivity to these uncertainties. The elicitation consisted of four workshops and two field trips to the Yucca Mountain area. Each panel member made an individual assessment or model of the igneous hazard on the basis of his or her interpretations of various probabilistic models. A logic tree approach was used to combine alternatives and to incorporate uncertainty. Each of the 10 experts' probability estimates was then combined with equal weight to produce a probability distribution of the annual frequency of intersection of a basaltic dike within the proposed repository footprint.

PSHA Expert Elicitation

SAR Section 2.2.2.1 described DOE's overall approach to developing a seismic hazard assessment for Yucca Mountain, including fault displacement hazards. This approach included an expert elicitation to develop a PSHA for Yucca Mountain (CRWMS M&O, 1998aa; BSC, 2004bj). DOE conducted its PSHA in the late 1990s using a methodology that DOE claims is consistent with a Level 4 expert elicitation as described in NUREG/CR–6372

(NRC, 1997aa). SAR Section 5.4.2 summarized DOE's bases and concluded that this methodology was also generally consistent with the nine-step procedure suggested in NUREG-1563 (NRC, 1996aa).

DOE's PSHA also followed the standard framework for PSHAs in using the recurrence curve approach (e.g., Cornell, 1968aa; McGuire, 1976aa). The basic elements of this framework are (i) identification and spatial distribution of seismic sources; (ii) characterization of each source in terms of its activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) description of ground motion attenuation relationships to model the distribution of the ground motions expected when a given magnitude earthquake occurs on a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, along with their associated uncertainties. Each logic tree pathway is intended to represent one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that DOE considers representative of the seismic hazard at a site, including variability and uncertainty.

To accomplish the PSHA, DOE hired two panels of experts. The first expert panel consisted of six three-member teams of geologists and geophysicists (seismic source teams) who developed probabilistic distributions to characterize relevant potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates, expressed along with estimates of their uncertainties, were specific to the regional crustal conditions of the western Basin and Range Province. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the PSHA.

Inputs from the expert teams were combined into a logic tree and the hazard computed using a modified version of the FRISK88 computer code (Risk Engineering, Inc., 1998aa). In the integration, DOE gave equal weight to all six source teams and seven ground motion experts. The resulting ground motion hazard curves express increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves include estimates of uncertainty.

The seismic source teams also developed a Probabilistic Fault Displacement Hazard Assessment as part of the PSHA. In that aspect of the PSHA elicitation, the experts derived probabilistic fault displacement hazard curves for nine demonstration points at or near Yucca Mountain. These demonstration points represent a range of faulting and related fault deformation conditions in the subsurface and near the sites of proposed surface facilities.

NRC staff's technical evaluation of the geological, geophysical, and seismological information used to support the expert elicitation are provided in TER Sections 2.2.1.2.2.3.2 and 2.1.1.1.3.5.2.

Saturated Zone Flow and Transport Expert Elicitation (SZEE)

SAR Section 2.3.9.2.2.6 discussed DOE's use of expert elicitation to address key issues associated with groundwater flow and transport in the saturated zone. SAR Section 5.4.3

summarized DOE's bases for asserting that SZEE was conducted in a manner generally consistent with the nine-step procedure suggested in NUREG–1563 (NRC, 1996aa).

In 1997, DOE carried out an expert elicitation to evaluate saturated zone flow and transport at the Yucca Mountain site (CRWMS M&O, 1998ab). The objective of SZEE was to quantify uncertainties associated with models and parameters key to modeling flow and transport in the saturated zone. A second objective was to reveal needed data collection and modeling that could reduce some of the more significant uncertainties. In this way, the expert elicitation was used to complement and guide data collection already underway, as well as to provide input to iterative performance assessment modeling by DOE.

Over 6 months, a panel of 5 experts in saturated zone hydrology was asked to address 16 technical issues related to the study of saturated zone groundwater flow and radionuclide transport at Yucca Mountain. DOE implemented many of the panel members' recommendations in subsequent site characterization activities. In particular, the panel recommended a range of values for vertical anisotropy, dispersivity, and specific discharge that DOE later used, along with other sources of information, to characterize the uncertainty of flow and transport of radionuclides beneath and downgradient of Yucca Mountain. Written elicitation summaries, prepared by each expert, were included in an appendix to the final elicitation report (CRWMS M&O, 1998ab).

NRC staff's technical evaluation of the geological, geophysical, and hydrological information used to support the expert elicitation, as well as of that information developed as a result of it, is provided in TER Volume 3, Section 2.2.1.3.8.

Staff Evaluation

The YMRP guidance for review of expert elicitation considers the nine-step procedure outlined in NUREG–1563 (NRC, 1996aa). These steps will be discussed in turn, with specific examples cited from the three elicitations, where appropriate.

Definition of Objectives (Step 1)

NRC staff notes that, in general, DOE defined specific objectives for each elicitation, on the basis of the descriptions of these objectives in the SAR (pp. 5.4-4, 5.4-7, and 5.4-10), discussion of the rationale for the elicitations in the respective elicitation reports, and direct observation by NRC staff members of the elicitation workshops and meetings.

In the PVHA elicitation, however, staff recognizes some shortcomings in the specific definition of objectives. Staff has previously documented, in NUREG–1762, Section 5.1.2.2.4.1, Igneous Activity (NRC, 2005aa), that a common definition of an igneous event or event class was not adequately specified at the beginning of the elicitation, and that these terms were not used consistently in the experts' probability models. Probability estimates for intrusive and extrusive events were not calculated separately, but were initially considered as a single probability by the experts. Because separate probability estimates needed to be developed for the DOE Total System Performance Assessment, DOE developed extrusive and intrusive probability estimates subsequent to the 1996 PVHA without re-engaging the experts to seek their opinions.

Selection of Experts (Step 2)

NRC staff notes that DOE generally followed published guidance in selecting experts. This is based on review of the criteria DOE used to select experts, as described in SAR Section 5.4; professional information provided about each expert in the elicitation reports; and NRC staff members' direct observations of the open, frank, and detailed technical discussion among the experts at the elicitation workshops.

For the PVHA, NRC staff notes that the 10 experts possessed the necessary knowledge and expertise and showed their ability to apply their knowledge and expertise. All 10 experts were identified in the 1996 PVHA report, and each expert's judgments were clearly documented. As was identified in its 1999 Issue Resolution Status Report (NRC, 1999aa), NRC staff believes that a greater balance of panel experts would have encompassed a wider range of viewpoints. NRC staff attributes DOE's failure to achieve this balance, in part, to the fact that some of the experts invited by DOE declined to participate as panel members. Also, subsequent to PVHA, NRC staff suggested that DOE strive for more thorough documentation of the expert selection processes and identify sources of potential bias and conflicts of interest (Austin, 1997aa, 1996aa).

For the PSHA and the SZEE, the NRC staff notes that the experts possessed the necessary knowledge and expertise. NRC staff also notes that the assembled experts for the two elicitations collectively represent an appropriately broad spectrum of the larger seismology and hydrology communities.

All of the final elicitation reports identified the participating subject matter experts, included summaries of their input to the elicitations, and provided rationales for their respective opinions. As DOE stated in SAR Section 5.4, the experts were not asked directly to disclose potential conflicts of interest, but each expert provided sufficient information about his or her past and current affiliations to satisfy the intent of the guidance in NUREG–1563.

Refinement of Issues (Step 3) and Assembly and Dissemination of Basic Information (Step 4)

On the basis of its direct observation and review, NRC noted that the geological, geophysical, hydrological, and seismological information DOE made available to each panel provided an adequate technical basis to support the three elicitations. During the early workshops and field trips, the experts developed lists of the most important subissues. This helped organize and focus the discussions in later elicitation workshops. Among the numerous subissues that the PVHA experts identified were structural control of igneous activity in the vicinity of Yucca Mountain, the quality and reliability of available age dating, and the selection of relevant natural analogues. DOE divided the PSHA into two panels of experts, each with its own set of experts. One panel focused on description and characterization of seismic sources, while the other panel focused on ground motion attenuation and modeling. Within the seismic source panel, DOE further developed three-person elicitation teams, composed of experts with varied expertise in geology, seismology, geophysics, and Basin and Range tectonics. Among the key subissues the experts identified in the SZEE were the causes and implications of the large hydraulic gradient, spatial distribution of flow, and the range of uncertainty in groundwater specific discharge.

Pre-Elicitation Training (Step 5)

On the basis of documentation provided in the elicitation reports as well as direct observations by NRC staff at the elicitation workshops, NRC staff notes that the subject matter experts received appropriate pre-elicitation training, consistent with NRC guidance/NUREG–1563. Experts received training on the elicitation process during the first workshop of each elicitation, as well as during subsequent workshops, including presentations on topics such as probability encoding, quantifying uncertainty, and identifying sources of bias.

Most of the workshops were held with sufficient advanced notice so that members of the public, affected parties, and NRC staff could directly observe the discussions among the experts and supporting technical teams. Many of the workshops included presentations by subject matter experts, both from within the teams or external to the elicitation. At later workshops, the experts presented their preliminary interpretations in a discussion format that allowed them to receive direct feedback from other experts or expert teams. Each of the elicitation projects included at least one field trip that allowed the experts to directly observe many of the important geologic features in the Yucca Mountain region. These field trips included discussions with subject matter experts and generalists on specific field investigations carried out on behalf of DOE in support of the site characterization. DOE provided meeting summaries of all the workshops in the elicitation reports.

Elicitation of Judgments (Step 6)

Upon completion of the workshops and field trips, facilitation teams, comprising generalists and normative experts, conducted comprehensive interviews of the experts to elicit their inputs that included discussion of how the information would be represented in the logic tree format used to calculate the results. These interviews were conducted expert by expert or, where applicable, team by team, and followed up with written documentation of the inputs.

Post-Elicitation Feedback (Step 7)

As documented in the elicitation reports and SAR Section 5.4, the experts were provided with both informal and formal feedback at many of the workshops. At least one workshop in each elicitation was dedicated to feedback and included initial sensitivity studies provided by the facilitation team to quantify the initial expert interpretations and, through sensitivity studies, to show which inputs had the greatest impact on the overall results. NRC staff noted that this aspect of DOE’s elicitation process met the criteria for timely feedback. However, DOE did not require the experts or expert teams to document the rationale for any changes made to their assessments after the feedback session. As stated in SAR Section 5.4 and in DOE (2009gn), DOE stated that this requirement could anchor the experts to their initial interpretations. DOE asserted that the experts would thus be reluctant to revise their interpretations after receiving feedback because doing so would also require them to provide full justification for the change. DOE also stated that its approach, in this regard, is consistent with the guidance contained in NUREG/CR–6372 (NRC, 1997aa). The NRC staff considers that, in this one respect, DOE’s approach is more comparable to that described in NUREG/CR–6372, in that the guidance in NUREG/CR–6372 does not specify that experts document the rationale for changes made to their assessment during or after feedback sessions.

Treatment of Disparate Views and Aggregation of Judgments (Step 8)

For all three elicitations, equal weighting was used to aggregate the elicited results. In the case of PSHA, results were aggregated giving equal weights to the inputs from the source teams, and equal weights to the ground motion models from the individual ground motion experts. In the other cases, equal weight was assigned to the results from each expert. The elicitation reports provided summaries of each expert's (or source team's) input, including sensitivity information to demonstrate the impact each expert or each source team's interpretations had on the final result.

Documentation (Step 9)

NRC staff notes that DOE properly documented all three elicitations. The elicitation reports provided comprehensive records of each elicitation, with the noted exception being formal documentation of individual experts' reasons for revising their interpretations during the elicitation process. DOE explained its rationale for this deviation in SAR Section 5.4. As stated previously, the NRC staff notes that DOE selected an approach that in this regard is similar to that of NUREG/CR–6372 (NRC, 1997aa).

The YMRP guidance for review of expert elicitation also considers the documentation and methodology used in updates to an elicitation. DOE chose not to update the PSHA or the SZEE. DOE did, however, reconvene the PVHA elicitation in 2004 to consider new information and to rely on a consistent set of event definitions and extrusive scenarios. Members of the NRC staff attended the public PVHA-Update (PVHA-U) workshops as observers. DOE published the results from the updated PVHA, or PVHA-U, after it submitted the SAR (SNL, 2008ah).

DOE did not directly use the PVHA-U results in its SAR or in direct support of models or parameters in the Total System Performance Assessment (TSPA). Estimates of the probability of igneous activity in the TSPA are based solely on the original 1996 PVHA. In a letter providing the PVHA-U report to NRC (Boyle, 2008aa), DOE characterized the PVHA-U results as information that supports the results of the 1996 PVHA. While the NRC staff notes that the results are similar, the underlying technical bases for the two elicitations are not the same. Because DOE referred to the PVHA-U results as confirming the 1996 PVHA results, the staff reviewed the PVHA-U report. On the basis of staff's subsequent review and direct observation of the PVHA-U workshops, the NRC staff noted that the PVHA-U was conducted in a manner generally consistent with the procedure suggested in NUREG–1563 (NRC, 1996aa). For these reasons, NRC staff notes that the PVHA-U was adequately documented and used appropriate elicitation methods, consistent with NUREG–1563.

NRC staff's technical evaluation of DOE's estimates of igneous event probability as they relate to the PVHA-U is given in TER Section 2.2.1.2.2.

2.5.4.4 NRC Staff Conclusions

The NRC staff notes that DOE adequately explained how expert elicitation was used consistent with the applicable guidance in NUREG–1563 and the YMRP.

2.5.4.5 References

Austin, J.H. 1997aa. "Documenting and Disclosing Potential Conflict of Interest in Expert Elicitations for the Geologic Repository Program." Letter (January 7) to R.A. Milner, DOE. Washington, DC: NRC.

Austin, J.H. 1996aa. "Implementation of NUREG–1563 in Expert Elicitations in Yucca Mountain, Nevada, Site Characterization Programs." Letter (December 31) to R.A. Milner, DOE. Washington, DC: NRC.

Boyle, W.J. 2008aa. "Transmittal of Report: Probabilistic Volcanic Hazard Analysis Update (PVHA-U) for Yucca Mountain, Nevada." Letter (October 17) to Director, DHLWRS, NRC. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

BSC. 2004bj. "Technical Basis Document No. 14: Low Probability Seismic Events." Rev. 1. MOL 20000510.0175. Las Vegas, Nevada: Bechtel SAIC Company, LLC.

Cornell, C.A. 1968aa. "Engineering Seismic Risk Analysis." *Bulletin of the Seismological Society of America*. Vol. 58. pp. 1,583–1,606.

CRWMS M&O. 1998aa. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada." WBS 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O. 1998ab. "Synthesis of Volcanism Studies for the Yucca Mountain Site Characterization Project." 3781MR1. MOL 19981207.0393. Las Vegas, Nevada: CRWMS M&O.

CRWMS M&O. 1996aa. "Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada." BA0000000–01717–2200–00082. Rev. 0. Las Vegas, Nevada: CRWMS M&O.

DOE. 2009av. DOE/RW–0573, "Safety Analysis Report Yucca Mountain Repository License Application." Rev. 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management.

DOE. 2009gn. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 5.1), Safety Evaluation Report Vol. 4, Chapter 2.5.1, Set 1." Letter (February 6) J.R. Williams to B.J. Benney (NRC). ML090371019. Washington, DC: DOE, Office of Technical Management.

McGuire, R.K. 1976aa. "FORTRAN Computer Program for Seismic Risk Analysis." USGS Open-File Report 76-67. Reston, Virginia: U.S. Geological Survey.

NRC. 2005aa. NUREG–1762, "Integrated Issue Resolution Status Report." Rev. 1. Washington, DC: NRC.

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. Washington, DC: NRC.

NRC. 1999aa. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Rev. 2. Washington, DC: NRC.

NRC. 1997aa. NUREG/CR–6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." Washington, DC: NRC.

NRC. 1996aa. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC.

Risk Engineering, Inc. 1998aa. *FRISK88 User Manual, Version 2.0*. Boulder, Colorado: Risk Engineering, Inc.

SNL. 2008ah. "Probabilistic Volcanic Hazard Analysis Update (PVHA U) for Yucca Mountain, Nevada." Rev. 01. Las Vegas, Nevada: Sandia National Laboratories.

(Intentionally Left Blank)

CHAPTER 21

Conclusions

The U.S. Nuclear Regulatory Commission staff has reviewed the Safety Analysis Report and the other information the U.S. Department of Energy (DOE) submitted. NRC staff notes that (i) the repository design is composed of multiple barriers and (ii) the results of the performance assessments for individual protection, human intrusion, and separate groundwater protection calculations are consistent with applicable YMRP guidance. In particular, the staff notes the following.

Multiple Barriers

- The repository includes multiple barriers, consisting of both natural barriers and engineered barrier systems.

- The proposed engineered barrier systems can be expected to work in combination with the natural barriers, consistent with the criteria for a system of multiple barriers.

- DOE identified the design features of the engineered barrier system and natural barrier features of the geologic setting that are considered barriers important to waste isolation, described the capability of the barriers identified as important to waste isolation (to isolate waste), and provided the technical basis for the description of the capability of the barriers that is consistent with the technical basis for the performance assessment evaluations.

Performance Assessments

- The performance assessment evaluations used to estimate repository performance include the appropriate features, events, and processes. DOE has provided an appropriate basis for the features, events, and processes that have been excluded from the performance assessment analysis.

- DOE appropriately considered events that have at least 1 chance in 100 million per year of occurring (i.e., igneous events, seismic events, and early failure of waste packages and drip shield events) in its performance assessment analyses.

- The model abstractions used in the performance assessment evaluations included applicable data related to the natural systems and the engineered barrier systems, appropriate consideration of uncertainty and variability in parameters and models, consideration of alternative models, and technical bases in support of the model abstractions. DOE should confirm its approach for decay chain radionuclide behavior by providing, through its performance confirmation program, information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone, as identified in TER Section 2.2.1.3.9.3.

- DOE has provided a reasonable basis for the statistical stability of the quantitative results of the Total System Performance Assessment (TSPA).

DOE submitted information consistent with the guidance in the Yucca Mountain Review Plan (YMRP). The technical approach and results in DOE's TSPA, including the average annual dose values and the performance of the repository barriers, as discussed in this volume are reasonable. DOE should confirm its approach for decay chain radionuclide behavior by providing, through its performance confirmation program, information to reduce uncertainty related to the likelihood of excess Po-210 occurring in the saturated zone, as identified in TER Section 2.2.1.3.9.3.

CHAPTER 22

Glossary

This glossary is provided for information and is not exhaustive. Terms shown in *italics* are included in this glossary.

absorption: The process of taking up by capillary, osmotic, solvent, or chemical action of molecules (e.g., absorption of gas by water) as distinguished from *adsorption*.

abstracted model: A model that reproduces, or bounds, the essential elements of a more detailed process model and captures *uncertainty* and v*ariability* in what is often, but not always, a simplified or idealized form. See *abstraction*.

abstraction: Representation of the essential components of a process model into a form suitable for use in a *total system performance assessment*. A model abstraction is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and *uncertainty* analyses.

adsorb: To collect a gas, liquid, or dissolved substance on a surface as a condensed layer.

adsorption: The adhesion by chemical or physical forces of molecules or ions (as of gases or liquids) to the surface of solid bodies. For example, the transfer of solute mass, such as *radionuclides*, in *groundwater* to the solid geologic surfaces with which it comes in contact. The term *sorption* is sometimes used interchangeably with this term.

advection: The process in which solutes, particles, or molecules are transported by the motion of flowing fluid.

aging: The retention of *commercial spent nuclear fuel* on the surface in *dry storage* to reduce its thermal output as necessary to meet proposed repository thermal management goals.

airborne mass loading: The amount of fine particulates resuspending above a surface deposit, generally expressed as mass per unit volume of air.

aleatory uncertainty: An *uncertainty* associated with the chance of occurrence of a feature, event, or process of a physical system or the environment such as the timing of a volcanic event. Also referred to as irreducible *uncertainty* because no amount of knowledge will determine whether or not a chance event will or will not occur. See also *epistemic uncertainty*.

Alloy 22: A nickel-based, *corrosion*-resistant alloy containing approximately 22 weight percent chromium, 13 weight percent molybdenum, and 3 weight percent tungsten as major alloying elements. This alloy is used as the outer container material in U.S. Department of Energy’s waste package design.

alluvium: Detrital (sedimentary) deposits made by flowing surface water on river beds, flood plains, and alluvial fans. It does not include subaqueous sediments of seas and lakes.

alternative: In the context of system analysis, plausible interpretations or designs that use assumptions other than those used in the base case, which could also be applicable or reasonable given the available scientific information. When propagated through a quantitative tool such as performance assessment, alternative interpretations can illustrate the significance of the *uncertainty* in the base case interpretation chosen to represent the system's probable behavior.

ambient: Undisturbed, natural conditions, such as ambient temperature caused by climate or natural subsurface thermal gradients, and other surrounding conditions.

anisotropy: Variation in physical properties when measured in different directions. For example, in layered rock, permeability is often greater within the horizontal layers than across the horizontal layers.

annual frequency: The number of occurrences of an event in 1 year.

aqueous: Pertaining to water, such as aqueous phase, aqueous species, or aqueous *transport*.

aquifer: A saturated underground geologic formation of sufficient permeability to transmit *groundwater* and yield water of sufficient quality and quantity to a well or spring for an intended beneficial use.

ash: Fragments of volcanic rock that are broken during an explosive volcanic eruption to less than 2 mm [0.08 in] in diameter. See also *tephra* and *pyroclastic.*

ash flow tuff: A type of volcanic rock formed by the deposition and accumulation of dominantly ash-size particles during an explosive eruption. Ash flows (also called *pyroclastic* flows) commonly result from eruptions of more viscous, silica-rich *magma* such as *rhyolite*. This rock type forms the host horizons for the proposed geologic repository at Yucca Mountain. See also *tuff* and *welded tuff*.

basalt: A common type of *igneous* rock that forms black, rubbly-to-smooth-surfaced lavas and black-to-red *tephra* deposits (frequently used as "lava rock" for barbecues).

borosilicate glass: A predominantly noncrystalline, relatively homogenous glass formed by melting silica and boric oxide together with other constituents such as alkali oxides. Borosilicate glass is a high-level radioactive waste material in which boron takes the place of the lime used in ordinary glass mixtures.

boundary condition: For a model, the establishment of a set condition for a given *variable,* often at the geometric edge of the model. An example is using a specified *groundwater* flux for *net infiltration* as a boundary condition for an *unsaturated zone flow* model.

bound: An analysis or selection of parameter values that yields limiting results, such that any actual result is certain to exceed these limits only with an extremely small likelihood.

breach: A penetration in the waste package caused by failure of the outer and inner containers or barriers that allows the *spent nuclear fuel* or the high-level radioactive waste to be exposed to the external environment and may eventually permit *radionuclide* release.

burnup: A measure of nuclear reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission, or as the amount of energy produced per unit weight of fuel.

burnup credit: The concept of taking credit for the reduction in reactivity (ability to undergo fission) due to fuel irradiation. The reduction in reactivity is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides.

caldera*:* A volcanic depression in the Earth's surface more than 1 km [0.7 mi] wide, formed by the collapse of the upper crust into an evacuated *magma* chamber during or after a large volcanic eruption. Many calderas resulting from the explosive eruption of large amounts of *rhyolite magma* are several tens of kilometers [up to 20 mi] wide.

calibration: (1) Comparison of model results with actual data or observations, and adjusting model parameters to increase the precision and/or accuracy of model results compared to actual data or observations. (2) For tools used for field or lab measurements, the process of taking instrument readings on standards known to produce a certain response, to check the accuracy and precision of the instrument.

canister: An unshielded cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level radioactive waste. It may serve as (i) a pour mold and container for vitrified high-level radioactive waste; (ii) a container for loose or damaged fuel rods, nonfuel components and assemblies, and other debris containing *radionuclides*; or (iii) a container that provides *radionuclide* confinement. Canisters are used in combination with specialized overpacks that provide structural support, shielding, or confinement for storage, transportation, and emplacement. Overpacks used for transportation are usually referred to as *transportation casks*; those used for emplacement in a proposed repository are referred to as waste packages.

carbon steel: A steel made with carbon up to about 2 weight percent and only residual quantities of other elements. Carbon steel is a tough but ductile and malleable material that is used in some components in U.S. Department of Energy's design of the engineered barrier system.

cask*:* (1) A heavily shielded container used for the dry storage or shipment (or both) of radioactive materials such as *spent nuclear fuel* or other high-level radioactive waste. Casks are often made from lead, concrete, or steel. Casks must meet regulatory requirements and are not intended for long-term disposal in a proposed repository. (2) A heavily shielded container that the U.S. Department of Energy would use to transfer *canisters* between waste handling facilities at the proposed repository.

cinder cone*:* A steep, conical hill formed by the accumulation of *ash* and coarser erupted material around a volcanic vent. Synonymous with scoria cone.

cladding: The metal outer sheath of a fuel rod generally made of a zirconium alloy, and in the early nuclear power reactors of stainless steel, intended to protect the uranium dioxide pellets, which are the nuclear fuel, from dissolution by exposure to high temperature water under operating conditions in a reactor. Often referred to as "clad."

climate: Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region.

climate states: Representations of *climate* conditions.

colloid: As applied to *radionuclide* migration, colloids are large molecules or very small particles, having at least one dimension with the size range of 10^{-6} to 10^{-3} mm [10^{-8} to 10^{-5} in] that are suspended in a solvent. Colloids in *groundwater* arise from clay minerals, organic materials, or (in the context of a proposed geologic repository) from corrosion of engineered materials.

commercial spent nuclear fuel: Nuclear fuel rods, forming a fuel assembly, that have been removed from a nuclear power plant after reaching the specified *burnup*.

conceptual model: A set of qualitative assumptions used to describe a system or subsystem for a given purpose. Assumptions for the model are compatible with one another and fit the existing data within the context of the given purpose of the model.

conduit: A pathway along which *magma* rises to the surface during a volcanic eruption. Conduits are usually cylindrical and flare upwards toward the surface vent. Conduits are near-surface features and develop along *dikes*, focusing *magma* flow from the longer and possibly narrower *dike* to the vent.

consequence: A measurable or calculated outcome of an event or process that, when combined with the probability of occurrence, gives a measurement of risk.

conservative: A condition of an analysis or a parameter value such that its use provides a pessimistic result, which is worse than the actual result expected.

corrosion: The deterioration of a material, usually a metal, as a result of a chemical or electrochemical reaction with its environment.

coupled processes: A representation of the interrelationships between *processes* such that the effects of variation in one process are accurately propagated among all interrelated *processes*.

crevice corrosion: *Localized corrosion* of a metal surface at, or immediately adjacent to, an area that is shielded from full exposure to the environment because of close proximity between the metal and the surface of another material.

criticality: The condition in which a fissile material sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.

diffusion: (1) The spreading or dissemination of a substance caused by concentration gradients. (2) The gradual mixing of the molecules of two or more substances because of random thermal motion.

diffusive transport: Movement of solutes because of their concentration gradient. Diffusive transport is the process in which substances carried in *groundwater* move through the subsurface by means of *diffusion* because of a concentration gradient.

dike: A tabular, generally vertical body of *igneous* rock that cuts across the *structure* of adjacent rocks. Dikes *transport* molten rock from depth to an erupting volcano.

dimensionality: Modeling in one, two, or three dimensions.

direct exposure: The manner in which an individual receives dose from being in close proximity to a source of radiation. Direct exposures present an external dose pathway.

dispersion (hydrodynamic dispersion): (1) The tendency of a solute (substance dissolved in *groundwater*) to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid were to move it. The tortuous path the solute follows through openings (pores and fractures) causes part of the dispersion effect in the rock. (2) The macroscopic outcome of the actual movement of individual solute particles through a porous medium. Dispersion dilutes solutes, including *radionuclides*, in *groundwater.*

disposal canister: A cylindrical metal receptacle designed to contain *spent nuclear fuel* and high-level radioactive waste as an integral part of the waste package.

disruptive event: An unlikely, off-normal event that, in the case of the proposed repository, could include volcanic activity, *seismic* activity, and nuclear *criticality*. Disruptive events alter the normal or likely behavior of the system.

dissolution: Dissolving a substance in a solvent.

distribution: In a *total system performance assessment*, the overall scatter of values for a specific set of numbers (e.g., *corrosion* rates, values used for a particular parameter, dose results). A term used synonymously with frequency distribution or probability distribution function. Distributions have structures that are the *probability* that a given value occurs in the set.

drift: From mining terminology, a horizontal underground passage. In the proposed Yucca Mountain repository design, drifts include excavations for emplacement (emplacement drifts) and access (access mains).

drift degradation: The progressive accumulation of rock rubble in a *drift* created by weakening and collapse of drift walls in response to stress from heating or earthquakes.

drip shield: A metallic structure placed along the extension of the emplacement *drifts* and above the *waste packages* to prevent *seepage* water from directly dripping onto the waste package outer surface. The drip shield may also prevent the *drift* ceiling rocks (e.g., due to *drift* spallation) from falling on the waste package.

dry storage: Storage of *spent nuclear fuel* without immersion of the fuel in water for cooling or shielding; it involves the encapsulation of spent fuel in a steel cylinder that might be in a concrete or massive steel *cask* or structure.

effective porosity: The fraction of a porous medium volume available for fluid flow and/or solute storage, as in the saturated zone. Effective porosity is less than or equal to the total void space (porosity).

empirical: Reliance on observation or experimentation rather than on a theoretical understanding of fundamental processes.

emplacement drift: See *drift*.

enrichment: The act of increasing the concentration of fissile isotopes from their value in natural uranium. The enrichment (typically reported in atom percent) is a characteristic of nuclear fuel.

eolian: Relating to processes caused by near-surface winds.

epistemic uncertainty: A *variability* that is due to a lack of knowledge of quantities or processes of the system or the environment. Also referred to as reducible *uncertainty*, because the state of knowledge about the exact value of a quantity or process can increase through testing and data collection. See also *aleatory uncertainty.*

equilibrium: The state of a chemical system in which the phases do not undergo any spontaneous change in properties or proportions with time; a dynamic balance.

events: In a *total system performance assessment*, (1) occurrences of phenomena that have a specific starting time and, usually, a duration shorter than the time being simulated in a *model*. (2) Uncertain occurrences of phenomena that take place within a short time relative to the time frame of the model.

event tree*:* A modeling tool that illustrates the logical sequence of events that follow an initiating event.

expected annual dose: The average annual radiological dose calculated for the reasonably maximally exposed individual, which includes the likelihood of the individual receiving a dose from all relevant exposure scenarios.

expert elicitation: A formal, highly structured, and well-documented process whereby expert judgments, usually of multiple experts, are obtained.

Exploratory Studies Facility: An underground laboratory at Yucca Mountain that includes a 7.9-km [4.9-mi] main loop (tunnel); a 2.8-km [1.75-mi] cross drift; and a research alcove system constructed for performing underground studies during site characterization.

extrusive (extrusion)*:* In relation to *igneous* activity, an event where *magma* erupts at the surface. An extrusion is the deposit formed by an extrusive event. See also *intrusive*.

fault (geologic): A planar or gently curved *fracture* across which there has been displacement parallel to the *fracture* surface.

features: Physical, chemical, thermal, or temporal characteristics of the site or proposed repository system. For the purposes of screening features, *events*, and *processes* for the total system performance assessment, a feature is defined to be an object, structure, or condition that has a potential to affect disposal system performance.

finite element analysis: A commonly used numerical method for solving mathematical equations in a variety of areas (e.g., hydrology, mechanical deformation). A technique in which algebraic equations are used to approximate the partial differential equations that comprise mathematical models to produce a form of the problem that can be solved on a computer. For this type of approximation, the area being modeled is formed into a grid with irregularly shaped blocks. This method provides an advantage in handling irregularly shaped boundaries (e.g., internal features such as faults) and surfaces of engineered materials. Values for parameters are frequently calculated at nodes for convenience, but are defined everywhere in the blocks by means of interpolation functions.

fissure: In relation to *igneous* activity, a fissure is an elongate vent or line of vents, formed when a *dike* breaks to the surface to start a volcanic eruption.

flow: The movement of a fluid such as air, water, or *magma*. Flow and *transport* are processes that can move *radionuclides* from the proposed repository to the receptor group location.

flow pathway: The subsurface course that water or a solute (and dissolved material) would follow in a given *groundwater* velocity field, governed principally by the hydraulic gradient.

fluvial: Processes related to the downslope movement of water on the Earth's surface.

fracture: A planar discontinuity in rock along which loss of cohesion has occurred. It is often caused by the stresses that cause folding and faulting. A *fracture* along which there has been displacement of the sides relative to one another is called a *fault*. A fracture along which no appreciable movement has occurred is called a joint. Fractures may act as fast paths for *groundwater* movement.

fragility*:* Fragility of a structure, system, or component is defined as the conditional probability of its failure, given a value of the ground motion, or response parameter, such as stress, bending moment, and spectral acceleration.

frequency: The number of occurrences of an observed or predicted event during a specific time period.

galvanic: Pertains to an electrochemical process in which two dissimilar electronic conductors are in contact with each other and with an electrolyte, or in which two similar electronic conductors are in contact with each other and with dissimilar electrolytes.

galvanic corrosion: Accelerated corrosion of a metal resulting from electrical contact with a more noble metal or nonmetallic conductor in a corrosive electrolyte.

geochemical: The distribution and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere; the movement of the elements in nature on the basis of their properties.

geophysics (geophysical survey; geophysical magnetic survey)*:* The study of the physical properties of rocks and sediment and interpretation of data derived from measurements made. Properties commonly measured are the velocity of sound (*seismic* waves) in rocks, density, and magnetic character. A program of measurements made on a series of rocks is usually termed a survey.

groundwater: Water contained in pores or *fractures* in either the unsaturated or saturated zones below ground level.

half-life: The time required for a radioactive substance to lose half of its activity due to radioactive decay. At the end of one half-life, 50 percent of the original radioactive material has decayed.

heterogeneity: The condition of being composed of parts or elements of different kinds. A condition in which the value of a parameter varies over the space an entity occupies, such as the area around the proposed repository, or with the passage of time.

hydrologic: Pertaining to the properties, distribution, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

igneous: (1) A type of rock that has formed from a molten, or partially molten, material. (2) A type of activity related to the formation and movement of molten rock, either in the subsurface (*intrusive*) or on the surface (*extrusive*).

infiltration: The process of water entering the soil at the ground surface. Infiltration becomes percolation when water has moved below the depth at which evaporation or transpiration can return it to the atmosphere. See also *net infiltration*.

intrusive (intrusion)*:* In relation to *igneous* activity, an event where *magma* approaches the surface but does not break through in an eruption. An intrusion is the solidified rock formed below the surface by an intrusive event. See also *extrusive*.

invert: A constructed surface that would provide a level *drift* floor and enable emplacement and support of the waste packages.

lithophysal: Containing lithophysae, which are holes in *tuff* and other volcanic rocks. One way lithophysae are created is by the accumulation of volcanic gases during the formation of the tuff.

localized corrosion: Corrosion at discrete sites (e.g., pitting and *crevice corrosion*).

magma: Molten or partially molten rock that is naturally occurring and is generated within the Earth. Magma may contain crystals along with dissolved gasses.

mathematical model: A mathematical description of a *conceptual model*.

matrix: Rock material and its pore space exclusive of *fractures*.

matrix diffusion: The process by which molecular or ionic solutes, such as *radionuclides* in *groundwater*, move from areas of higher concentration to areas of lower concentration. For the proposed Yucca Mountain repository, this process refers to the movement of radionuclides by diffusion between the *fracture* and *matrix* continua.

matrix permeability: The capability of the *matrix* to transmit fluid.

mean (arithmetic): For a statistical data set, the sum of the values divided by the number of items in the set. The arithmetic average, sometimes referred to as expected value.

mechanical disruption: Damage to the *drip shield* or waste package because of external forces.

median: A value such that one-half of the observations are less than that value and one-half are greater than the value.

meteorology: The study of climatic conditions such as precipitation, wind, temperature, and relative humidity.

microbe: An organism too small to be viewed with the unaided eye. Examples of microbes are bacteria, protozoa, and some fungi and algae.

microbially influenced corrosion: Deterioration of metals as a result of the metabolic activity of microorganisms.

migration: *Radionuclide* movement from one location to another within the engineered barrier system or the environment.

mineralogical: Of or relating to the chemical and physical properties of minerals, their occurrence, and their classification.

model: A depiction of a system, phenomenon, or process, including any hypotheses required to describe the system or explain the phenomenon or process.

model support: A process used to gain confidence in the reasonableness of *model* results through comparison with outputs from detailed process-level models and/or empirical observations such as laboratory tests, field investigations, and natural analogues.

natural analogues: Observable *features, events,* or *processes*, which provide insights on similar features, events, or processes that are difficult to observe in the proposed repository system.

near-field: The area and conditions within the proposed repository including the *drifts* and waste packages and the rock immediately surrounding the *drifts*. The near-field is the region in and around the proposed repository where the excavation of the proposed repository *drifts* and the emplacement of waste have significantly impacted the natural *hydrologic* system.

net infiltration: The downward flux of infiltrating water that escapes below the zone of evapotranspiration. The bottom of the zone of evapotranspiration generally coincides with the lowermost extent of plant roots.

nominal scenario class: The *scenario*, or set of related scenarios, that describes the expected or nominal behavior of the system as perturbed only by the presence of the proposed repository. The nominal scenarios contain all likely *features, events*, and *processes* that have been retained for analysis.

numerical model: An approximate representation of a mathematical model that is constructed using a numerical description method such as finite volumes, finite differences, or finite elements. A numerical model is typically represented by a series of program statements that are executed on a computer.

occupational dose*:* The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include doses received from background radiation, from any medical administration the individual has received, from exposure to individuals who were administered radioactive material and released under 10 CFR 35.75, from voluntary participation in medical research programs, or as a member of the public (10 CFR 20.1003, "Occupational dose").

oxidation: A *corrosion* reaction in which the corroded metal forms an oxide, usually applied to reaction with a gas containing elemental oxygen, such as air.

parameter: Data, or values, such as those that are input to computer codes for a *total system performance assessment* calculation.

patch: In the U.S. Department of Energy modeling of waste package *corrosion*, a patch is the minimal surface area of the waste package over which uniform corrosion occurs, as opposed to localized corrosion in *pits*.

pathway: A potential route by which *radionuclides* might reach the accessible environment and pose a threat to humans. For example, direct exposure is a human external pathway, and inhalation and ingestion are human internal pathways.

permeability: A measure of the ease with which a fluid such as water or air moves through a rock, soil, or sediment.

phase: A physically homogeneous and distinct portion of a material system, such as the gaseous, liquid, and solid phases of a substance. In liquids and solids, single phases may coexist.

phase stability: A measure of the ability of a particular phase to remain without transformation.

pit: A small cavity formed in a solid as a result of *localized corrosion*.

pitting corrosion: *Localized corrosion* of a metal surface, confined to a small area, that takes the form of cavities (*pits*).

porosity: The ratio of the volume occupied by openings, or voids, in a soil or rock, to the total volume of the soil or rock. Porosity is expressed as a decimal fraction or as a percentage.

probabilistic: Based on or subject to *probability*.

probability: The chance that an outcome will occur from the full set of possible outcomes. Knowledge of the exact probability of an event is usually limited by the inability to know, or compile, the complete set of possible outcomes over time or space.

probability distribution: The set of outcomes (values) and their corresponding probabilities for a random *variable*. See *distribution*.

processes: Phenomena and activities that have gradual, continuous interactions with the system being modeled.

process model: A depiction or representation of a *process*, along with any hypotheses required to describe or to explain the *process*.

pyroclastic: In relation to *igneous* activity, this describes fragments or fragmental rocks and deposits produced by explosive volcanic activity, where the *magma* is ripped apart during the release of gas and/or by interaction with surface and near-surface water.

quaternary*:* The period of geologic time from about 2.6 million years ago to the present day.

radioactive decay: The process in which one *radionuclide* spontaneously transforms into one or more different *radionuclides*, which are called daughter *radionuclides*.

radioactivity: The property possessed by some elements (such as uranium) of spontaneously emitting energy in the form of radiation as a result of the decay (or disintegration) of an unstable atom. Radioactivity is also the term used to describe the rate at which radioactive material emits radiation.

radiolysis: Chemical decomposition by the action of radiation.

radionuclide: An unstable isotope of an element that decays or disintegrates spontaneously, thereby emitting radiation. Approximately 5,000 natural and artificial radioisotopes have been identified.

range (statistics): The numerical difference between the highest and lowest value in any set.

receptor: An individual for whom radiological doses are calculated or measured.

redistribution: Mobilization and transport of surface deposits by wind and water.

reliability*:* The *probability* that the item will perform its intended function(s) under specified operating conditions for a specified period of time.

repository footprint: The outline of the outermost locations of where the waste is proposed to be emplaced in the proposed Yucca Mountain repository.

retardation: Slowing or stopping *radionuclide* movement in *groundwater* by mechanisms that include *sorption* of *radionuclides*, *diffusion* into rock *matrix* pores and microfractures, and trapping of particles in small pore spaces or dead ends of microfractures.

rhyolite*:* A common type of *igneous* rock that forms light-colored, rough blocky-surfaced lavas and white-grayish-yellow *tephra* deposits. A common fragment type is pumice. Rhyolitic *magma* has a high viscosity, and the resulting lava flows are usually quite short and thick. It more frequently erupts explosively from the volcano and forms *ash-flow tuffs*.

risk: The probability that an undesirable event will occur, multiplied by the consequences of the undesirable event.

risk assessment: An evaluation of potential *consequences* or hazards that might be the outcome of an action, including the likelihood that the action might occur. This assessment focuses on potential negative impacts on human health or the environment.

risk informed, performance based: A regulatory approach in which *risk* insights, engineering analysis and judgments, and performance history are used to (i) focus attention on the most important activities; (ii) establish objective criteria on the basis of *risk* insights for evaluating performance; (iii) develop measurable or calculable parameters for monitoring system and licensee performance; and (iv) focus on the results as the primary basis for regulatory decision making.

rockfall: The release of *fracture*-bounded blocks of rock from the *drift* wall, usually in response to an earthquake.

rock matrix: See *matrix*.

runoff: Lateral movement of water at the ground surface, such as down steep hillslopes or along channels, that is not able to infiltrate at a specified location.

scenario: A well-defined, connected sequence of *features, events*, and *processes* that can be thought of as an outline of a possible future condition of the proposed repository system. Scenarios can be undisturbed, in which case the performance would be the expected, or nominal, behavior for the system. Scenarios can also be disturbed, if altered by disruptive events such as human intrusion or natural phenomena such as *volcanism* or nuclear *criticality*.

scenario class: A set of related *scenarios* sharing sufficient similarities that they can usefully be aggregated for screening or analysis. The number and breadth of scenario classes depend on the resolution at which scenarios have been defined.

scoria cone: See *cinder cone*.

seepage: The inflow of *groundwater* moving in *fractures* or *matrix* pores of permeable rock to an open space in the rock. For the proposed Yucca Mountain repository, seepage refers to water dripping into a *drift*.

seismic: Pertaining to, characteristic of, or produced by earthquakes or Earth vibrations.

seismic hazard curve: A graph showing the ground motion *parameter* of interest, such as peak ground acceleration, peak ground velocity, or spectral acceleration at a given frequency, plotted as a function of its annual *probability* of exceedance.

seismic performance: Seismic performance of *structures, systems, and components* refers to their ability to perform intended safety functions during a *seismic* event, expressed as the annual probability of exceeding a specified limit condition (stress, displacement, or collapse). This is also referred to as the probability of failure, or probability of unacceptable performance, P_F.

sill: A tabular, generally flat-lying body of *intrusive igneous* rock that lies along (is concordant with) the *structure* of adjacent rocks. Sills are part of the *transport* system for molten rock (*magma*) rising from depth to the surface. See also *dike*.

sorb: To undergo a process of *sorption*.

sorption: The binding, on a microscopic scale, of one substance to another. Sorption is a term that includes both *adsorption* and *absorption* and refers to the binding of dissolved *radionuclides* onto geologic solids or waste package materials by means of close-range chemical or physical forces. Sorption is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the material they encounter along the *flow* path.

sorption coefficient (K_d): A numerical means to represent how strongly one substance *sorbs* to another.

source term: Types and amounts of *radionuclides* that are the source of a potential release.

spatial variability: A measure of how a property, such as rock *permeability*, varies at different locations in an object such as a rock formation.

speciation: The existence of the elements, such as *radionuclides*, in different molecular forms in the aqueous phase.

spent nuclear fuel: Nuclear reactor fuel that has been used to the extent that it can no longer effectively sustain a chain reaction and that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. This fuel is more radioactive than it was before irradiation and releases significant amounts of heat from the decay of its fission product *radionuclides*.

stainless steel: A class of iron-base alloys containing a minimum of approximately 10 percent chromium to provide *corrosion* resistance in a wide variety of environments.

stratigraphy: The branch of geology that deals with the definition and interpretation of rock strata; the conditions of their formation, character, arrangement, sequence, age, and distribution; and especially their correlation by the use of fossils and other means of identification. See *stratum*.

stratum: A layer of rock or soil with geologic characteristics that differ from the layers above or below it.

stress corrosion cracking: A cracking process that requires the simultaneous action of a corrosive substance and sustained (residual or applied) tensile stress. Stress corrosion cracking excludes both the fracture of already corroded sections and the localized corrosion processes that can disintegrate an alloy without the action of residual or applied stress.

structure: In geology, the arrangement of the parts of geologic features or areas of interest such as folds or *faults*. This includes features such as fractures created by faulting, and joints caused by the heating of rock. For engineering usage, see *structures, systems, and components*.

structures, systems, and components: A *structure* is an element, or a collection of elements, that provides support or enclosure, such as a building, *aging* pad, or *drip shield*. A *system* is a collection of components, such as piping; cable trays; conduits; or heating, ventilation, and air conditioning equipment, that are assembled to perform a function. A *component* is an item of mechanical or electrical equipment, such as a *canister* transfer machine, transport and emplacement vehicle, pump, valve, or relay.

tectonic: Pertaining to geologic *features* or *events* created by deformation of the Earth's crust.

tephra: A collective term for all clastic (fragmental) materials ejected from a volcano during an eruption and transported through the air.

thermal chemical: Of or pertaining to the effect of heat on chemical conditions and reactions.

thermohydrologic: Of or pertaining to changes in *groundwater* movement due to the effects of changes in temperature.

thermal mechanical: Of or pertaining to changes in mechanical properties from effects of changes in temperature.

total system performance assessment: A *risk assessment* that quantitatively estimates how the proposed Yucca Mountain repository system will perform in the future under the influence of specific *features, events*, and *processes*, incorporating *uncertainty* in the models and *uncertainty* and *variability* of the data.

transparency: The ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results. A logical structure ensures completeness and facilitates in-depth review of the relevant issues. Transparency is achieved when a reader or reviewer has a clear picture of what was done in the analysis, why it was done, and the outcome.

transpiration: The removal of water from the ground by vegetation (roots).

transport: A process that allows substances such as contaminants, *radionuclides*, or *colloids*, to be carried in a fluid from one location to another. Transport processes include the physical mechanisms of *advection*, convection, *diffusion*, and *dispersion* and are influenced by the chemical mechanisms of *sorption*, leaching, precipitation, *dissolution*, and complexation.

tuff: A general term for volcanic rocks that formed from rock fragments and *magma* that erupted from a volcanic vent, flowed away from the vent as a suspension of solids and hot gases, or fell from the eruption cloud, and consolidated at the location of deposition. Tuff is the most abundant type of rock at the Yucca Mountain site.

uncertainty: How much a calculated or measured value varies from the unknown true value. See also *aleatory uncertainty* and *epistemic uncertainty.*

unsaturated zone flow: The movement of water in the unsaturated zone, as driven by capillary, viscous, gravitational, inertial, and evaporative forces.

variable: A nonunique property or attribute.

variability (statistical): A measure of how a quantity varies over time or space.

volcanism: Pertaining to extrusive *igneous* activity.

wash: In a relation to landforms, a streambed, dry or running, usually in an arid environment.

watershed: The area drained by a river system including the adjacent ridges and hillslopes.

welded tuff*:* A *tuff* deposited under conditions where the particles that make up the rock remained sufficiently hot to cohere. In contrast to nonwelded tuff, welded tuff is denser, less porous, and more likely to be fractured (which increases *permeability*).