DOC.20080208.0001

QA: QA

ANL-DS0-NU-000001 REV 00

February 2008

Screening Analysis of Criticality Features, Events, and Processes for License Application

NOTICE OF OPEN CHANGE DOCUMENTS - THIS DOCUMENT IS IMPACTED BY THE LISTED CHANGE DOCUMENTS AND CANNOT BE USED WITHOUT THEM.
--
1) ACN-001, DATED 03/10/2008

THIS DOCUMENT IS IMPACTED BY THE FOLLOWING "ERROR RESOLUTION DOCUMENTS", WHICH SHOULD BE REVIEWED PRIOR TO USING THIS DOCUMENT:
--
1) ANL-DS0-NU-000001 ERD 01 - ACCESSION NO. LLR.20080401.0255
2) ANL-DS0-NU-000001 ERD 02 - ACCESSION NO. LLR.20080423.0161

Prepared for:
U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Office of Repository Development
1551 Hillshire Drive
Las Vegas, Nevada 89134-6321

Prepared by:
Sandia National Laboratories
OCRWM Lead Laboratory for Repository Systems
1180 Town Center Drive
Las Vegas, Nevada 89144

Under Contract Number:
DE-AC04-94AL85000

QA: QA

Screening Analysis of Criticality Features, Events, and Processes for License Application

ANL-DS0-NU-000001 REV 00

February 2008

INTENTIONALLY LEFT BLANK

Scientific Analysis/Calculation Signature Page/Change History

1. Total Pages: 140

Complete only applicable items.

2. Document Title

Screening Analysis of Criticality Features, Events, and Processes for License Application

3. DI (Including Revision No. and Addendum No.)

ANL-DS0-NU-000001 REV 00

	Printed Name	Signature	Date
4. Originator	John A. McClure	John A McClure	02/06/2008
5. Checker	James K. Knudsen	James K Knudsen	02/06/2008
6. QCS/Lead Lab QA Reviewer	Brian T. Mitcheltree	Brian Mitcheltree	2/6/08
7. Responsible Manager/Lead	John C. Wagner	John Scaglione FOR J.C. Wagner	2/6/08
8. Responsible Manager	Clifford L. Howard	Cliff Howard	2/6/08

9. Remarks

This document supersedes entirely *Probability of Postclosure Criticality*, CAL-MGR-NU-000012, and *Screening Analysis of Criticality Features, Events, and Processes for License Application*, ANL-EBS-NU-000008.

Change History

10. Revision No. and Addendum No.	11. Description of Change
00	Initial Issue. Upon completion, this analysis report either addresses or resolves Condition Reports CR-5600, CR-5757, and CR-8396 associated with this or prior revisions. CR-5600: ANL-EBS-NU-000008 was reviewed as part of an Extent of Condition associated with the Condition Description. No action items were generated for ANL-EBS-NU-000008. CR-5757: ANL-EBS-NU-000008 cites DTN: LB0304SMDCREV2.002 and DTN: LB0307SEEPDRCL.002 that are cited in the CR Condition Description. No action items were generated for ANL-EBS-NU-000008. CR-8396: Action Number 8396-009 is to revise the DIRS and/or CAL-EBS-NU-000012 to evaluate the use of the unqualified DTN: MO0502SPACOMPA.000 as a direct input. Action Number 8396-009 was resolved by citing the qualified DTN: ~~MO0508SPACOMPA.002~~ MO0501BPVELEMP.001 as necessary in this revision.

JAM 02/06/2008

CH 02/06/08

INTENTIONALLY LEFT BLANK

CONTENTS

Page

CONTENTS (Continued)

Page

FIGURES

INTENTIONALLY LEFT BLANK

TABLES

Page

INTENTIONALLY LEFT BLANK

ACRONYMS

BWR	boiling water reactor
DOE	U.S. Department of Energy
EBS	engineered barrier system
FEPs	features, events, and processes
FFTF	Fast Flux Test Facility
HLW	high-level (radioactive) waste
k_{eff}	effective neutron multiplication factor
NRC	U.S. Nuclear Regulatory Commission
OCB	outer corrosion barrier
PGV	peak horizontal ground velocity
PWR	pressurized water reactor
RST	residual (tensile) stress threshold
SCC	stress corrosion cracking
SNF	spent nuclear fuel
TAD	transportation, aging, and disposal (canister)
TSPA-LA	total system performance assessment for the license application
TWP	technical work plan
YMP	Yucca Mountain Project
YMRP	Yucca Mountain Review Plan

INTENTIONALLY LEFT BLANK

1. PURPOSE

The purpose of this analysis report is to evaluate the features, events, and processes (FEPs) associated with criticality and document the screening decision for either inclusion or exclusion of criticality in the Total System Performance Assessment for License Application (TSPA-LA). The FEPs associated with criticality address scenarios that include initiators of sequences of events or processes that could lead to configurations that have potential for criticality in the repository. Thus, criticality is a single event and a screening decision, either *Included* or *Excluded* for all criticality FEPs collectively, is based on the total probability of occurrence of configurations with potential for criticality for the repository rather than evaluating each FEP independently. The technical basis for each individual FEP is summarized in the conclusions for this analysis. This information is required by the U. S. Nuclear Regulatory Commission (NRC) as documented in 10 CFR Part 63 ([DIRS 180319], Section 102(j)) and § 63 (proposed rule) (70 FR 53313 [DIRS 178394]). Proposed amendments to the 10 CFR Part 63 to address a dose standard after 10,000 years are given in *Implementation of a Dose Standard After 10,000 Years* (70 FR 53313 [DIRS 178394]). The approach used for estimating the probability of postclosure criticality resulting from events occurring under conditions ranging from early failure of engineered barriers to disruptive geological environments makes use of various processes and tools for identifying potentially critical configurations (including probability of occurrence) and calculating the maximum effective neutron multiplication factor (k_{eff}) of configurations, if necessary.[1] The methodology provides a means to evaluate potential postclosure criticality events for the following conditions and locations:

1) The full range of waste form conditions (intact, degraded, and degradation products)

2) For postulated conditions of impairment to the engineered systems (waste package and other engineered barriers)

3) For the range of possible locations (in-package, near-field, and far-field where the near-field is the region inside the drift excluding the waste package and the far-field is the volume outside the drift).

An evaluation of the criticality FEP scenarios[2] from the configuration generator model report (BSC 2004 [DIRS 172494], Section 7) for configurations with potential for criticality has identified two dominant leitmotivs common to each of the in-package scenarios in sequences of events that must occur for a criticality event to be credible. The two independent events are: (1) improper fabrication, resulting in the absence and/or loss of efficacy of the neutron absorber material and, (2) for pressurized water reactor (PWR) spent nuclear fuel (SNF), improper loading of fuel assemblies. These events are captured in the configuration generator model report (BSC 2004 [DIRS 172494], Figure I-5) as Top Events "NA-MISLOAD" (neutron absorber-misload) and "WF-MISLOAD" (waste form-misload). The estimated probability of occurrence for the

---

[1] If the probability of occurrence of a configuration with potential for criticality is below the low probability screening criterion (10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53319, Section 342(a)) (proposed rule), reactivity calculations are not necessary for screening purposes.

[2] *FEP scenarios* refers to scenarios that are based on 16 FEPs associated with initiating events and locations affecting the criticality potential in the repository.

sequence of events that include these two top events is central to the FEPs screening justifications for the in-package scenarios. Likewise, evaluation of the criticality FEP scenarios for the near- and far-field locations resulted in the identification of the accumulation of a critical fissile mass as a common event central to the FEPs screening justifications for the external scenarios. A number of other events also belong to these and similar sequences of events but are not quantified in this analysis (e.g., seepage distribution and climatic variations). Since the probabilities of events in a sequence cannot exceed one, quantifying additional events cannot result in an increased higher probability for the sequence. The probability estimates for the occurrence of configurations with criticality potential from the FEPs screening analysis provide conservative estimates for the probability of criticality for the various scenarios associated with early failure of engineered barriers as well as disruptive geological scenarios (i.e., seismic, rockfall, and igneous). The justification for the values being conservative estimates of the probability of criticality for the scenarios is further developed in Section 6.2.

The FEPS screening analysis is an essential part of the overall postclosure criticality analysis methodology documented in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) and is performed in accordance with the methodology. The basis for the postclosure criticality analysis methodology is the master scenario list (YMP 2003 [DIRS 165505], Section 3.3), which identifies possible degraded configurations resulting from the set of initiating events. Degradation scenarios are built from criticality FEPs as described in § Section 3.3. These include scenarios that lead to potentially critical configurations inside the waste package, outside of the waste package in the near field environment, and outside the waste package in the far field environment. *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) also includes a description of how these configurations are grouped into standard classes to make the problem manageable, while also ensuring that a comprehensive set of configurations is considered.

The FEPs screening analysis provides a method for evaluating whether the various configurations have potential for criticality and provides a means for identifying any configurations with such potential that cannot be screened from further analysis on the bases of low probability. The probabilities for dominant events identified in the individual criticality FEP scenarios that contribute to a potential criticality event are explicitly quantified and summed to determine an upper bound on the overall probability of criticality in the repository based on the probability of occurrence of configurations with potential for criticality. For other events and/or criticality FEP scenarios, the probabilities are not explicitly quantified as stated above. The result of this process is that a conservative value is determined for the overall probability of a criticality in the repository that is sufficiently low to allow criticality to be screened from consideration in the TSPA-LA for 10,000 years after closure with respect to the regulatory probability criterion (10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53319, Section 342(a)) (proposed rule). This evaluation is the basis for the screening justifications for the criticality FEPs in Section 7.

The analyses in this document do not consider Naval Nuclear Propulsion Program SNF.

1.1 PLANNING AND DOCUMENTATION

The FEPs screening approach used in this analysis deviates from the approach described in Section 2.1.8 of the technical work plan (TWP) entitled *Technical Work Plan for Postclosure Criticality* (SNL 2007 [DIRS 178869]). The TWP entailed using *Configuration Generator Model* (BSC 2004 [DIRS 172494]) and *Criticality Model* (BSC 2004 [DIRS 168553]) along with SAPHIRE software (V. 7.18 STN: 10325-7.18-00 [DIRS 160873]) to perform the probability calculations. Use of these models and software was predicated on using an event tree/fault tree methodology for performing the probability calculations. A review of the scenarios indicated that the probability of criticality is controlled by the values associated with only a few events as stated in Section 1 and that the overall probability of criticality could be shown to be below the screening criterion without performing a full scenario development and analysis. Thus, a bound on the probability of criticality was estimated using simple calculations in the analysis and not with the event tree/fault tree methodology from the configuration generator model. (Note: A more detailed analysis of one seismic event sequence resulted in a probability for the event sequence significantly below the value from a simplified approach (Section 6.4.2.1).)

This analysis also deviates from the TWP in that the NUREG-1804 (NRC 2003 [DIRS 163274]) acceptance criteria specified in *Technical Work Plan for Postclosure Criticality* (SNL 2007 [DIRS 178869], Table 3) have not all been specifically addressed with respect to how this analysis satisfies the particular criterion. Several of these criteria (model and design criteria) were determined to be not relevant to this document as noted in Section 4.2.1. All of the relevant criteria listed in Table 3 of *Technical Work Plan for Postclosure Criticality* (SNL 2007 [DIRS 178869]) are included in Table 4.2-1 with the method of addressing relevant criteria described.

Documentation requirements for this analysis report are described in the TWP (SNL 2007 [DIRS 178869], Section 8.4).

1.2 SCOPE

The scope of this report is to describe, evaluate, and document screening decisions for the disposition of criticality FEPs for TSPA-LA and the technical bases for these decisions. For any FEP excluded from the TSPA-LA, analyses are required to provide justification for the TSPA-LA disposition, based on the various supporting technical analysis reports and model reports that (collectively) provide justification for the FEP disposition. This analysis report provides a screening justification for exclusion of the criticality FEPs from the TSPA-LA on the basis of low probability and discusses the technical bases that support that decision. It also provides appropriate references to Yucca Mountain Project (YMP) and non-YMP information that support the decision for exclusion of the FEPs. The criticality FEPs (Table 1.2-1) are also screened together as a unit. The probability of the criticality event class is the sum of the probabilities from all sixteen criticality FEPs. If this value is below the screening criterion (i.e., at least one chance in 10,000 of occurring over 10,000 years), then all contributors to this sum are below the screening criterion. If one FEP cannot be screened out (i.e., excluded), the entire event class would be screened in (i.e., included) since the overall probability of a criticality event would be above the screening threshold. In such a case, the scenarios would be reviewed to determine the scenario(s) that must be included in the TSPA-LA or the design changed to meet the screening criterion.

An overview of the YMP FEP scenario and analysis development process in *Features, Events, and Processes for the Total System Performance Assessment* (SNL 2007 [DIRS 179476], Sections 6.3 and 6.4) describes the TSPA-LA FEP identification and screening process that led to the development of FY 2007 LA FEP List and Screening (DTN: MO0706SPAFEPLA.001 [DIRS 181613]). Changes in FEP list, FEP names, and FEP descriptions can also be traced through that report. The criticality FEPs addressed in this report form a subset of FY 2007 LA FEP List and Screening (DTN: MO0706SPAFEPLA.001 [DIRS 181613]). These FEPs are listed in Table 1.2-1 by number in column 1, name in column 2, and description in column 3. Note that an "intact" waste package in this FEPs analysis includes "loss of containment but internal structures and waste form not degraded" as well as the package being sealed as at the time of repository closure. A loss of containment for waste packages includes any breach of the outer corrosion barrier (OCB) (e.g., from stress corrosion cracking (SCC), localized corrosion, or shearing). The sixteen criticality FEPs address scenarios derived from four initiating events (early failure of engineered barriers, seismic, rockfall, and igneous) in four environments (i.e., in-package intact, in-package degraded, external near-field, and external far-field). (The four environments cover only three locations as the in-package intact and degraded configurations for current designs differ primarily in the waste form composition.) While the engineered barrier system (EBS) components and neutron absorber materials are designed to maintain their function in nominal repository environments over the first 10,000-year period after repository closure by specifying a corrosion allowance or minimum thickness (SNL 2007 [DIRS 179394], Table 4-1, Items 03-07 and 03-10), disruptive environments must be considered as well as uncertainty in the corrosion rates, thus degraded states must also be considered.

Table 1.2-1. Criticality FEPs List Utilized in Screening Analysis

FEP Number	FEP Name	FEP Description
FEPs Associated with Nominal (Early Failure) Event Sequence Initiators		
2.1.14.15.0A	In-package criticality (intact configuration)	The waste package internal structures and the waste form remain intact. If there is a breach (or are breaches) in the waste package that allows water to either accumulate or flow-through the waste package, then criticality could occur in situ. In-package criticality resulting from disruptive events is addressed in separate FEPs.
2.1.14.16.0A	In-package criticality (degraded configurations)	The waste package internal structures and the waste form may degrade. If a potentially critical configuration (sufficient fissile material and neutron moderator present with a lack of neutron absorbers) develops, a criticality event could occur in situ. Potential in situ critical configurations are defined in Figures 3-2a and 3-2b of *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]). In-package criticality resulting from disruptive events is addressed in separate FEPs.
2.1.14.17.0A	Near-field criticality	Near-field criticality could occur if a fissile material-bearing solution from the waste package is transported into the drift and the fissile material is precipitated into a critical configuration. Potential near-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3a). Near-field criticality resulting from disruptive events is addressed in separate FEPs.

Table 1.2-1. Criticality FEPs List Utilized in Screening Analysis (Continued)

FEP Number	FEP Name	FEP Description
FEPs Associated with Seismic Event Sequence Initiators		
2.2.14.09.0A	Far-field criticality	Far-field criticality could occur if a fissile material-bearing solution from the waste package is transported beyond the drift and the fissile material is precipitated into a critical configuration. Potential far-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3b). Far-field criticality resulting from disruptive events is addressed in separate FEPs.
2.1.14.18.0A	In-package criticality resulting from a seismic event (intact configuration)	The waste package internal structures and the waste form remain intact either during or after a seismic disruptive event. If there is a breach (or are breaches) in the waste package that allows water to either accumulate or flow-through the waste package, then criticality could occur in situ.
2.1.14.19.0A	In-package criticality resulting from a seismic event (degraded configurations)	Either during or as a result of a seismic disruptive event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figures 3-2a and 3-2b).
2.1.14.20.0A	Near-field criticality resulting from a seismic event	Either during or as a result of a seismic disruptive event, near-field criticality could occur if fissile material-bearing solution from the waste package is transported into the drift and the fissile material is precipitated into a critical configuration. Potential near-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3a).
2.2.14.10.0A	Far-field criticality resulting from a seismic event	Either during or as a result of a seismic disruptive event, far-field criticality could occur if fissile material-bearing solution from the waste package is transported beyond the drift and the fissile material is precipitated into a critical configuration. Potential far-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3b).
FEP Number	**FEP Name**	**FEP Description**
FEPs Associated with Rockfall Event Sequence Initiators		
2.1.14.21.0A	In-package criticality resulting from rockfall (intact configuration)	The waste package internal structures and the waste form remain intact either during or after a rockfall event. If there is a breach (or are breaches) in the waste package that allows water to either accumulate or flow-through the waste package then criticality could occur in situ.
2.1.14.22.0A	In-package criticality resulting from rockfall (degraded configurations)	Either during or as a result of a rockfall event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figures 3-2a and 3-2b).
2.1.14.23.0A	Near-field criticality resulting from rockfall	Either during or as a result of a rockfall event, near-field criticality could occur if fissile material-bearing solution from the waste package is transported into the drift and the fissile material is precipitated into a critical configuration. Potential near-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3a).
2.2.14.11.0A	Far-field criticality resulting from rockfall	Either during or as a result of a rockfall event, far-field criticality could occur if fissile material-bearing solution from the waste package is transported beyond the drift and the fissile material is precipitated into a critical configuration. Potential near-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3a).

Table 1.2-1. Criticality FEPs List Utilized in Screening Analysis (Continued)

FEP Number	FEP Name	FEP Description
FEPs Associated with Igneous Event Sequence Initiators		
2.1.14.24.0A	In-package criticality resulting from an igneous event (intact configuration)	The waste package internal structures and the waste form remain intact either during or after an igneous disruptive event. If there is a breach (or are breaches) in the waste package that allows water to either accumulate or flow-through the waste package then criticality could occur in situ.
2.1.14.25.0A	In-package criticality resulting from an igneous event (degraded configurations)	Either during or as a result of an igneous disruptive event, the waste package internal structures and the waste form may degrade. If a critical configuration develops, criticality could occur in situ. Potential in situ critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figures 3-2a and 3-2b).
2.1.14.26.0A	Near-field criticality resulting from an igneous event	Either during or as a result of an igneous disruptive event, near-field criticality could occur if fissile material-bearing solution from the waste package is transported into the drift and the fissile material is precipitated into a critical configuration. Potential near-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3a).
2.2.14.12.0A	Far-field criticality resulting from an igneous event	Either during or as a result of an igneous disruptive event, far-field criticality could occur if fissile material-bearing solution from the waste package is transported beyond the drift and the fissile material is precipitated into a critical configuration. Potential far-field critical configurations are defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Figure 3-3b).

Source: DTN: MO0706SPAFEPLA.001 [DIRS 181613].

Scenarios important for criticality are discussed in Section 6 followed by the individual FEP (FEP number, name, and description) evaluations and screening decision (screening justification or TSPA-LA disposition) information. A summary of the scenario probabilities is given in Section 7.1 and the criticality FEPs screening justification in Section 7.3.

1.3 SCIENTIFIC ANALYSIS LIMITATIONS AND USE

This report is intended to provide screening information to supplement the YMP FEP database and provide source documentation of the screening justification for the criticality related FEPs. There are no restrictions on the subsequent use of this report; however, the following limitations apply:

- Because this analysis report cites YMP and non-YMP documents, the limitations of this report inherently include any limitations or constraints described in the cited documents.

- A canister-based waste package design is used for all commercial and DOE waste forms anticipated for disposal in the repository (SNL 2007 [DIRS 179394], Section 1; SNL 2007 [DIRS 179567], Section 4). This design has a number of variants to accommodate the particular canister styles and waste forms (i.e., the commercial SNF variant for the 21-PWR and 44-BWR (boiling water reactor) loading patterns (SNL 2007 [DIRS 179394], Section 4.1.1.2) with a lengthened variant for the 12-PWR loading pattern (SNL 2007 [DIRS 179394], Section 4.1.3), and three DOE-owned SNF variants (SNL 2007 [DIRS 179567], Table 4-4). Note that the various canisters are inserted into the waste package, and hence any reference to the general term "waste package" throughout this document refers to the complete system of a loaded SNF canister, high-level waste (HLW) canisters (for codisposal (CDSP) waste packages), waste package inner shell, and OCB. As a detailed design does not exist for the TAD canister to be used for the 21-PWR and 44-BWR variants, the criticality FEPs screening decision is based on analyses of surrogate systems. These surrogates are the 21-PWR site-specific canister/basket and the TAD canister performance specification (SNL 2007 [DIRS 179394], Section 1). The 12-PWR TAD canister waste package operational, fabrication, and loading characteristics are expected to be similar to those for the 21–PWR TAD canister waste package, thus the 12-PWR TAD canister design variant is included in the probability evaluations forming the basis for the FEPs screening decision for license application. The PWR loading curve is sufficiently robust to encompass all of the potential design variants over the 10,000 year regulatory period in order to accommodate multiple canister criticality control design configurations (SNL 2008 [DIRS 182788], Section 1), which include the 12-PWR TAD canister.

- For screening purposes, this analysis report generally uses mean values of probability distributions as a basis for reaching an include/exclude decision. Determinations of mean values are based on the range of possible values with a corresponding uncertainty range. However, certain data may only be provided as a median, most probable, or point value that is noted where used in the analysis.

- The screening justifications for the criticality FEPs are based, in part, on probabilities for the occurrence of undetected fabrication, closure, and/or assembly loading errors associated with canisters. The possibility of criticality for the various waste forms requires the presence of such errors, as canisters fabricated and loaded according to design specifications remain subcritical (Section 6.2). Generic processes, operations, and human error probabilities have been used in the analysis as surrogates to calculate the requisite probabilities as detailed processes have not been defined for the various canister operations. The use of generic reliability assessment values for evaluating operational and/or fabrication errors is expected to generate more limiting (i.e., higher probability of failure) results than are expected during actual fabrication and process operations performed by operators trained on those specific processes. Generic human reliability analyses, of necessity, include sample distributions in their studies of the performance of basic operations that are drawn from a general population. The human error factor information is often derived by extrapolating data from performance measures only marginally related the process being evaluated (Swain and Guttman 1983 [DIRS 139383], p. 1-6). Fabrication processes for QA or any other type of manufacturing control require training on the processes. In general, the rates of operational errors for such specialized processes are likely to be lower than rates from operations performed by generic populations. Thus, considered collectively, the probabilities generated in this analysis are expected to be conservative estimates for actual values.

- Fabrication activities for waste packages are to be performed in accordance with a quality control program as specified in Section 9.4 of *Waste Package Fabrication Specification* as cited in Section 4.1.2.1 of *Total System Performance Assessment Data Input Package for Requirements Analysis for Transportation Aging and Disposal Canister and Related Waste Package Physical Attributes Basis for Performance Assessment* (SNL 2007 [DIRS 179394]). Quality checks (e.g., weight measurements of loaded DOE-owned SNF canisters containing absorber material in shot form) in addition to quality control programs may be necessary to sufficiently minimize operational or process failures.

- The various surrogate processes and operations used in this analysis are typical of quality control procedures and are considered reasonable proxies for such quality control procedures. However, as the FEPs screening justification is based, in part, on the probabilities derived from these surrogates, results from analyses of the final operational and fabrication procedures must demonstrate that the overall error probabilities from the fabrication and operational processes satisfy the FEPs screening justifications.

- The overall probability of criticality estimate does not include an evaluation of Naval Nuclear Propulsion Program SNF. The overall probability for the repository will remain below the regulatory criterion provided that the value for the naval SNF is less than 6.3×10^{-5} for the repository over 10,000 years.

- The results of the FEP screening presented herein are specific to the repository design and processes for YMP available at the time of the TSPA-LA. Changes in direct inputs listed in Section 4.1, in license application postclosure design parameters used for this evaluation, or in other subsurface conditions will require evaluation to determine whether the changes are within the limits stated in the FEP evaluations. Engineering and design changes are subject to evaluation to determine whether there are any adverse impacts to safety, as codified at 10 CFR Part 63 ([DIRS 180319], Subpart 73 and Subparts F and G) (see also the requirements at 10 CFR Part 63 [DIRS 180319], Subpart 44).

1.4 IMPLEMENTATION OF DISPOSAL CRITICALITY ANALYSIS METHODOLOGY

The criticality FEPs screening analysis implements the risk-informed, performance-based disposal criticality analysis methodology as documented in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]). An overview of the disposal criticality analysis methodology is presented in Figure 1.4-1 (YMP 2003 [DIRS 165505], Figure 3-1). The text in various boxes in Figure 1.4-1 has been modified from the original figure to improve the clarity of the items. The shaded boxes in Figure 1.4-1 signify the portion of the methodology associated with the FEPs screening analysis. The development of potential criticality scenarios is based on the standard configuration classes of the master scenario list (Box 1 of Figure 1.4-1) (YMP 2003 [DIRS 165505], Section 3.3). These criticality scenarios have been identified as having the most likely potential to increase the maximum k_{eff} of an in-package or external system. The criticality FEPs screening decision is based on probabilities calculated in Section 6 to evaluate the potential criticality scenarios of the various waste package/waste form combinations.

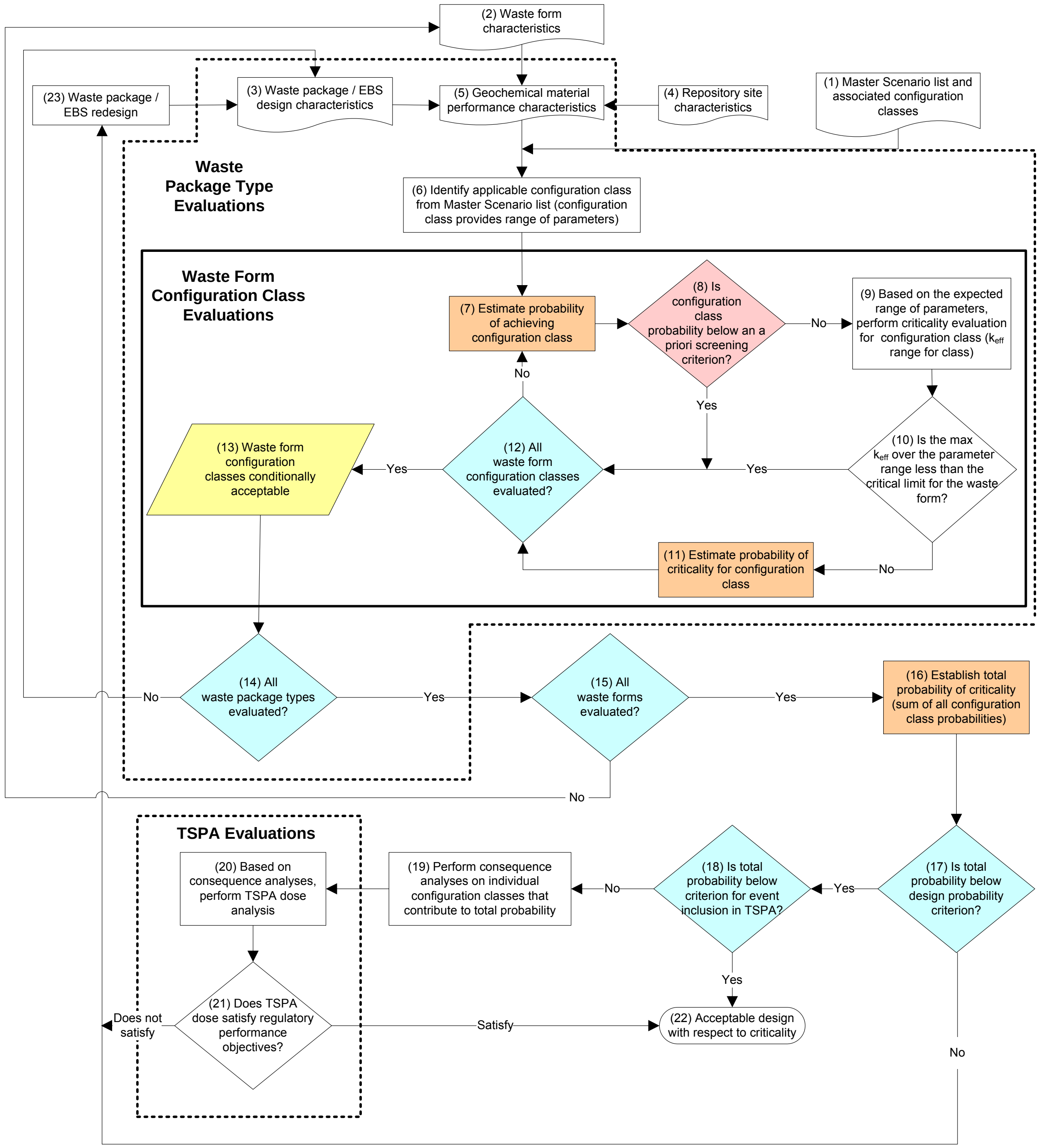
Note that the in-package intact configurations (Table 1.2-1) are not specifically identified in the master scenario list, because that list addresses degraded configurations (YMP 2003 [DIRS 165505], Section 3.3). The in-package intact configuration location addresses events such as canister and waste package OCB fabrication errors, neutron absorber misloads, and waste form misloads for configurations without degradation to evaluate those events for criticality potential and to include their probability in the overall criticality FEP screening probability.

Potential criticality scenarios applicable for the various waste package/waste form combinations are evaluated in a systematic manner using either cogent but non-quantitative arguments or an event tree methodology as documented in *Configuration Generator Model* (BSC 2004 [DIRS 172494]). The latter process is quantitative where the characteristics of the waste form, waste package, drip shield and repository (Boxes 2, 3, and 4 of Figure 1.4-1), as well as the geochemical performance characteristics (Box 5), are used to develop and define end states that represent the configuration classes derived from criticality scenarios (Boxes 6 and 7). The methodology focuses on evaluation of the probability of occurrence of configurations and, for configurations with potential for criticality, the probability of criticality for those configurations. A configuration class (a set of similar configurations whose composition and geometry are defined by specific parameters) is considered to have *potential for criticality* if the probability of the configuration class formation is above an a priori probability-screening criterion (Box 8) (i.e., an initial test of the likelihood for the occurrence of the configuration class). Having *potential for criticality* does not mean that a configuration is or even can be critical but that the configuration must be further analyzed to better quantify the probability of criticality for the configuration. The a priori probability screening criterion above is defined in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Section 3.2.1) to be well below (a minimum of two orders of magnitude) the screening criterion from 10 CFR Part 63 (Footnote 1). This criterion is used to screen from further consideration configuration classes that contribute insignificantly to the total probability of a criticality occurring in the repository during the 10,000-year period following closure of the repository where *insignificant* means an evaluation that the probability of the event would not change the overall result. The *probability of criticality* (Box 11), derived from the probability values for the range of configuration class parameters, is evaluated only for configuration classes for which the k_{eff} range exceeds the critical limit for the waste form (Box 10).

Once all waste package/waste form combinations are evaluated (Boxes 7 through 15), the probabilities from the individual waste package/waste form configuration classes are summed to obtain the total probability of criticality for the period defined in 10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53319, Section 342(a)) (proposed rule) (Box 16). The total probability of criticality is then compared to the design probability criterion (YMP 2003 [DIRS 165505], Section 3.2.3) of one chance of occurring during the initial 10,000-year time period following repository closure (Box 17). If the total probability of criticality is equal to or exceeds the design probability criterion, then a redesign of the waste packages or other components is necessary (Box 23) to reduce the total probability of criticality sufficiently to meet the design probability criterion.

If the estimated total probability of criticality is less than the design probability criterion, the total probability is then compared to the probability criterion for exclusion of events established in 10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53319, Section 342(a)) (proposed rule) as less than one chance in 10,000 of occurring during the 10,000-year period following repository closure (Box 18). If the total probability is less than this latter probability criterion, then the repository design is acceptable with respect to criticality concerns (Box 22). The criticality evaluations are then complete and criticality can be excluded from further evaluation in the TSPA. Otherwise, criticality consequence evaluations will be necessary (Box 19) for the development of additional radionuclide source terms for inclusion in the TSPA (Box 20).

The methodology outlined in Figure 1.4-1 permits the evaluation of the scenarios from the master scenario list (YMP 2003 [DIRS 165505], Section 3.3) as discussed in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) provided sufficient information about the scenarios is available. A review of the relevant configurations and analyses evaluated for criticality FEPs screening decisions and in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008 [DIRS 182788]) indicated that the overall probability of criticality is sensitive to only a few events. Additionally, a review of the FEPs scenarios associated with criticality indicated that the estimated probability of criticality could be shown to be below the regulatory screening criterion without performing a full scenario development and analysis. Thus, a conservative estimate of the probability of criticality can be calculated where the focus is placed on conditions necessary for criticality, and the additional probabilities associated with various environmental conditions and degradation mechanisms are not quantified, but would not exceed one and likely would be much less than one, resulting in a monotonically decreasing value of the overall probability for the sequence. Thus, quantitative evaluations are used in this analysis for estimating a conservative value for the probability of criticality in the repository where sufficient data are available and qualitative assessments used where sufficient data are not available. The conservative estimate of the probability of criticality is the basis for developing the FEP screening justifications.



Source: YMP 2003 [DIRS 165505], Figure 3-1.

Figure 1.4-1. Overview of Approach to the Disposal Criticality Analysis Methodology

INTENTIONALLY LEFT BLANK

2. QUALITY ASSURANCE

Technical Work Plan for: Postclosure Criticality (SNL 2007 [DIRS 178869], Section 8.1) determined that the development of this analysis report and the associated activities are subject to quality assurance requirements in accordance with YMP procedures. This report contributes to the analysis and modeling used to support postclosure safety and performance assessment. This report investigates the performance with respect to criticality concerns of the following systems, structures, and components that are important to waste isolation or important to barrier capability:

Commercial SNF Cladding
DOE and Commercial Waste Packages
Drip Shield
Saturated Zone (between the repository and the accessible environment)
Surface Topography, Soils and Bedrock (extrusive igneous scenario only)
Unsaturated Zone above the Repository
Unsaturated Zone below the Repository
Waste Form.

Although these barriers are categorized as "Safety Category" in *Q-List* (BSC 2005 [DIRS 175539], Table A-1), the evaluations and conclusions of this analysis report do not directly impact the features important to safety, defined in LS-PRO-0203, *Q-List and Classification of Structures, Systems, Components and Barriers*. The methods used to control the electronic management of data as required by IM-PRO-002, *Control of the Electronic Management of Information*, are identified in *Technical Work Plan for: Postclosure Criticality* (SNL 2007 [DIRS 178869], Section 8.4).

Also, in accordance with the TWP (SNL 2007 [DIRS 178869], Section 2.1.8), development of this analysis was controlled by SCI-PRO-005, *Scientific Analyses and Calculations*.

INTENTIONALLY LEFT BLANK

3. USE OF SOFTWARE

3.1 QUALIFIED AND BASELINE SOFTWARE

No baseline software was used in this analysis.

3.2 COMMERCIAL OFF-THE-SHELF SOFTWARE

Microsoft Excel® 2003 SP2, bundled with Microsoft Office 2003, is a commercial off-the-shelf software program used in this report. Microsoft Excel 2003 SP2 was installed on a Dell Celeron PC equipped with the Windows XP Version 2002 operating system and is appropriate for this application as it offers the mathematical and graphical functionality necessary to perform and document the numerical manipulations used in this report. The Excel computations performed in this report use only standard built-in functions and are documented in sufficient detail to allow an independent technical reviewer to reproduce or verify the results by visual inspection or hand calculation without recourse to the originator. The Excel files are included in the output DTN: MO0705CRITPROB.000. The calculation results are not dependent upon the use of this software; therefore, use of this software is not subject to IM-PRO-003, *Software Management*.

Mathcad® Version 14 (STN: 611161-14.0-00), which is commercially available off-the-shelf software, was installed on a DELL OptiPlex GX745 personal computer running Microsoft Windows XP Professional and used in the preparation of this report. Mathcad® is a problem-solving environment used in calculations and analyses to manipulate the inputs using standard mathematical expressions and operations. It is also used to tabulate and chart results. Standard functions of Mathcad® are used. The inputs and results are documented in sufficient detail to allow an independent repetition of computations. Thus, Mathcad® is used only as a worksheet and not as a software routine. Mathcad V. 14 is an exempt software product in accordance with IM-PRO-003, Section 2.

Inputs, outputs, and formulas used for the various Mathcad® calculations are documented in *Waste Package Flooding Probability Evaluation* (SNL 2008 [DIRS 184078], Appendix B). The electronic files for the calculations (Mathcad and printable copies) may be found in DTN: MO0712PBANLNWP.000 [DIRS 184480]. Note that if the Mathcad file is recalculated after opening, the seed must be set to the default value of one (1) to generate equivalent results (i.e., *Seed(1)*).

INTENTIONALLY LEFT BLANK

4. INPUTS

Technical product input usage is categorized in SCI-PRO-004, *Managing Technical Product Inputs*, as either direct input or indirect input. Direct input (addressed in this Section) is input used in a technical product that is directly relied on to support the results or conclusions. Indirect input is used to provide supporting information and is not used in the development of results or conclusions in the technical product. Supporting information for the direct input data is also provided in this section to aid in transparency and clarity.

All direct inputs used in this report are identified in Section 4.1. The direct inputs were obtained from controlled source documents and other appropriate sources in accordance with the controlling procedure SCI-PRO-004. The methods used for qualification of external data are discussed in Appendix II. The FEP screening criteria derived from 10 CFR Part 63 [DIRS 180319] and expanded in *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]) that are relevant to the FEP screening analysis are identified in Section 4.2 together with the method for addressing these criteria. Lastly, codes and standards applicable to the criticality FEP screening analysis are identified in Section 4.2.

4.1 DIRECT INPUTS

The following sections present the direct inputs used to perform the screening justifications for postclosure criticality FEPs which have been obtained from DTN: MO0706SPAFEPLA.001 [DIRS 181613] and listed in Table 1.2-1. Supporting information for the direct input data is also provided in this section for aiding transparency and clarity. Use of these data is justified as they are extracted from qualified project sources and their application is compatible with their developed purpose and limitations.

4.1.1 Mean Annual Seismic Exceedance Frequency Range and Time of Seismic Event

The mean annual seismic exceedance frequency of concern with respect to the probability of criticality evaluation ranges from 10^{-4} to 10^{-8} per year (SNL 2007 [DIRS 176828], Table 6-61). Seismic events occur randomly in time and are considered as independent events with regard to magnitude, time, and location. These events are modeled as a Poisson process (SNL 2007 [DIRS 176828], Section 5.2) that represents a compromise between the complexity of natural processes, the availability of information, and the sensitivity of relevant results. The range of annual exceedance frequencies used for particular seismic consequence evaluations may not cover the entire range due to varying thresholds for damage (e.g., the range for seismic faulting analyses is from approximately 10^{-7} to 10^{-8} per year) (SNL 2007 [DIRS 176828], Table 6-65).

4.1.2 Waste Package Fabrication and Operational Error Probabilities

As stated in Section 1, one of the principal events that can lead to configurations with potential for criticality is the Top Event “NA-MISLOAD” (BSC 2004 [DIRS 172494], Figure I-5) representing a neutron absorber misload in a canister. The neutron absorber misload event represents the improper performance of the neutron absorber plates due to fabrication related errors (e.g., incorrect material installed during fabrication, absorber content of plates outside specified range). An absorber misload event can only occur during fabrication of a canister or its components due to process or procedural errors and are similar to waste package and drip shield early failure mechanisms (SNL 2007 [DIRS 178765], Section 6.2). Errors in fabrication and operational processes are primarily due to human factors that are common to the various processes. Surrogate fabrication and operational processes with associated human factor errors have been evaluated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007 [DIRS 178765]) and results are used for such initiating events for the waste package and drip shield early failure mechanisms. The surrogate processes are:

1. Improper performance of the neutron absorber plates represented as a material selection error in the waste package component fabrication processes (SNL 2007 [DIRS 178765], Section 6.3.2)

2. Failure of the waste package and canister drying/inerting process represented as an operational process error (SNL 2007 [DIRS 178765], Section 6.3.5)

3. Drip shield misplacement allowing the possibility of advective seepage flow directly on a waste package OCB (SNL 2007 [DIRS 178765], Section 6.4.4)

4. Fabrication flaws allowing increased susceptibility to SCCs (SNL 2007 [DIRS 178765], Section 6.3).

Waste package fabrication and operational process error probabilities have been obtained from DTNs: MO0701PASHIELD.000 [DIRS 180508] and MO0705EARLYEND.000 [DIRS 180946]. The probability values assigned to absorber plate misloads due to material selection errors, waste package and canister operational process failures, waste package SCC mitigation process failures, and the occurrence of OCB closure lid weld flaws for this analysis are listed in Table 4.1-1. The operational process failures include the drying and inerting process and OCB outer lid weld stress mitigation process. These processes are conceptually similar as each requires operator actions and the human error failure rate from the OCB outer lid weld stress mitigation process is assigned to each one in Table 4.1-1.

Table 4.1-1. Undetected Errors in Waste Package Fabrication and Operational Processes

Waste Package Operations	Probability perCanister
Absorber material selection error [a]	1.25×10^{-7} per canister
Drying and inerting process failure	3.84×10^{-5} per canister
Outer closure lid weld stress mitigation process failure [a]	3.84×10^{-5} per canister
Emplacement error for drip shield [a]	4.36×10^{-9} per drip shield
Fraction of waste package OCB lid weld flaws oriented normally to surface [b]	8.0×10^{-3}
Probability of undetected fabrication defects in a waste package OCB [c]	1.13×10^{-4} per waste package
Probability of at least one flaw in waste package OCB lid closure weld [d]	1.56×10^{-1}

Sources: [a] DTN: MO0705EARLYEND.000 [DIRS 180946], file: *Table 1.doc*, Table 1.
[b] DTN: MO0701PASHIELD.000 [DIRS 180508], file: *Tables for DTN Readme.doc*, Table 1.
[c] DTN: MO0701PASHIELD.000 [DIRS 180508], file: *SAPHIRE OUTPUT.zip*.
[d] DTN: MO0701PASHIELD.000 [DIRS 180508], file: *EarlyFail-WeldDefects.zip*, Section A.7.

4.1.3 Waste Package Population

The percent breakdown of the anticipated emplacement inventory by waste package variant for the nominal 70,000 metric tons of heavy metal inventory limit established for disposal in the monitored geologic repository is given in Table 4.1-2 and in Output DTN: MO0705CRITPROB.000, file: *Waste Pkg Inventory.xls*. This inventory consists of a diverse collection of waste form designs and compositions with the DOE-owned SNF being the largest contributor to the waste form diversification. The strategy for applying the disposal criticality analysis methodology for the commercial SNF makes use of three basic waste package variants containing transportation, aging, and disposal canisters (i.e., a 21-PWR TAD variant, a 12-PWR Long TAD variant, and a 44-BWR TAD variant (DTN: MO0702PASTEAM.001 [DIRS 179925], spreadsheet: "COMMERCIAL," Item 6, cells J113 to K114). DOE-owned SNF canisters are codisposed with HLW in the codisposal waste package variants. The strategy for applying the methodology for the DOE-owned SNF was to categorize the divers forms into nine distinct groups identified in *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in the Repository* (DOE 2004 [DIRS 170071], Table A-2). The cross reference between the waste form type listed by Wheatley (2007 [DIRS 181533]) and that listed in Table A-2 (DOE 2004 [DIRS 170071]) is given in the notes for Table 4.1-2.

The DOE-owned SNF[3] inventory numbers in the point "Estimate" column (rounded up) from Wheatley (2007 [DIRS 181533]) are used for the postclosure criticality screening analysis as listed in Table 4.1-2 as these data supersede similar information from DTN: MO0702PASTREAM.001 [DIRS 179925], spreadsheet: "NONCOMMERCIAL." The inventory recommended by Wheatley (2007 [DIRS 181533]) is slightly (approximately 0.15%) greater than that from DTN: MO0702PASTREAM.001 [DIRS 179925], spreadsheet: "NONCOMMERCIAL." Note that this estimate is not necessarily bounding as the actual DOE-owned SNF inventory is uncertain (Wheatley 2006 [DIRS 179407]).

4.1.4 Probability of Igneous Events

The igneous disruptive event (intersection of the repository footprint by a volcanic dike or dike system) is described in *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada* (BSC 2004 [DIRS 169989]). The annual frequency of igneous disruptive events is characterized in the cited reference by a probability distribution having a mean value of 1.7×10^{-8} per year where the 5th and 95th percentiles are 7.4×10^{-10} per year and 5.5×10^{-8} per year, respectively (DTN: LA0307BY831811.001 [DIRS 164713], file: *Pecdist-la.xls*, spreadsheet: "Table 22"). The mean frequency value corresponds to a probability of 1.7×10^{-4} for the regulatory period of 10,000 years following closure of the repository (40 CFR Part 197 [DIRS 184076], Subpart B, Section 36).

[3] DTN: MO0702PASTREAM.001 [DIRS 179925], spreadsheet: "NON-COMMERCIAL," lists a DOE-owned SNF inventory that is lower by five waste packages than the inventory recommended by Wheatley (2007 [DIRS 181533]).

Table 4.1-2. Breakdown of Emplacement Inventory by Waste Package Variant

Waste Package Sequence	Waste Package Variant	Number of Waste Packages [a]	Fraction of Total Inventory	Number of Waste Packages by Type	Fraction of Total Inventory by Type	Number of Waste Packages	Fraction of Total Inventory
1	21-PWR TAD [b]	4,402	0.4088	4,568	0.4243	7,483	0.6950
2	12-PWR Long TAD [c]	166	0.0154				
3	44-BWR TAD [d]	2,915	0.2707	2,915	0.2707		
6	DOE1-Long [e, r]	128	0.0119	143	0.0133	3,284	0.3050
7	DOE1-Short [e,q]	15	0.0014				
8	DOE2-Short [f, q]	89	0.0083	89	0.0083		
9	DOE3-Long [g, r]	2	0.0002	17	0.0016		
10	DOE3-Short [g, q]	15	0.0014				
11	DOE3-MCO [g, s]	201	0.0187	201	0.0187		
12	DOE4-Long [h r]	70	0.0065	733	0.0681		
13	DOE4-Short [h, q]	663	0.0616				
14	DOE5-Long [i, r]	40	0.0037	53	0.0049		
15	DOE5-Short [i, q]	13	0.0012				
16	DOE6-Long [j, r]	570	0.0529	572	0.0531		
17	DOE6-Short [j, q]	2	0.0002				
18	DOE7-Long [k, r]	236	0.0219	991	0.0920		
19	DOE7-Short [k, q]	755	0.0701				
20	DOE8-Long [l, r]	8	0.0007	18	0.0017		
21	DOE8-Short [l, q]	10	0.0009				
22	DOE9-Long [m, r]	420	0.0390	458	0.0425		
23	DOE9-Short [m, q]	38	0.0035				
24	DOE9-MCO [m, s]	9	0.0008	9	0.0008		
Totals		10,767 [t]	1.00	10,767	1.00	10,767	1.00

Sources: [a] Commercial Inventory (DTN: MO0702PASTREAM.001 [DIRS 179925], spreadsheet: "COMMERCIAL," Item 6, cells J113 to K114);
DOE-owned SNF Inventory labeled as Cx, x = 1-9 (Wheatley 2007 [DIRS 181533], Point Estimate column {rounded up});

NOTES:
[b] 21-PWR TAD – 21-PWR TAD canister waste package variant.
[c] 12-PWR Long TAD – 12-PWR Long TAD canister waste package variant.
[d] 44-BWR TAD – 44-BWR canister waste package variant.
[e] DOE1 – Mixed Oxide (MOX) DOE-owned SNF C2; representative fuel type – Fast Flux Test Facility (FFTF).
[f] DOE2 – Uranium-Zirconium Hydride (UZrHx) DOE-owned SNF C7; representative fuel type – TRIGA.
[g] DOE3 – Uranium Metal (U-Metal) DOE-owned SNF C1; representative fuel type – N Reactor.
[h] DOE4 – High-Enriched Uranium Oxide (HEU Oxide) DOE-owned SNF C4; representative fuel type – Shippingport pressurized water reactor.
[i] DOE5 – Uranium/Thorium Oxide (U/Th Oxide) DOE-owned SNF C5; representative fuel type – Shippingport light water breeder reactor.
[j] DOE6 – Uranium/Thorium Carbide (U/Th Carbide) DOE-owned SNF C6; representative fuel type – Fort St. Vrain.
[k] DOE7 – Aluminum-based DOE-owned SNF C8; representative fuel type – Advanced Test Reactor.
[l] DOE8 – Uranium-Zirconium/Uranium-Molybdenum (U-Zr/U-Mo) Alloy DOE-owned SNF C3; representative fuel type – Enrico Fermi.
[m] DOE9 – Low-Enriched Uranium Oxide (LEU Oxide) DOE-owned SNF C9; representative fuel type – Three Mile Island II (TMI II).
[q] Codisposal Short waste package variant.
r Codisposal Long waste package variant.
[s] Codisposal MCO waste package variant.
[t] Naval Nuclear Propulsion Program SNF not included in the total.

MCO = multicanister overpack.

Given that an igneous disruptive event occurs in the vicinity of the repository, there is an estimated 0.28 probability (SNL 2007 [DIRS 177432], Table 7-1) of at least one extrusive center forming within the repository boundary. This translates to a mean frequency of ($1.7 \times 10^{-8} \times 0.28 =$) 4.8×10^{-9} per year for development of at least one extrusive center (i.e., conduit) in the repository.

The marginal distribution for the number of extrusive conduits that may be formed during the disruptive igneous scenario ranges from zero through 13, with one being the most probable number (DTN: LA0307BY831811.001 [DIRS 164713], file: *Pecdist-la.xls*, spreadsheet: "Table 19").

Basaltic magma is transported from a region of melting in the earth's mantle to the earth's surface through dikes. In the Yucca Mountain region, dikes are typically 1 to 12-m wide (DTN: LA0612DK831811.001, [DIRS 179987], file: *LA0612DK831811_001*, spreadsheet: EPAR_TPO-13Jan07).

Any drip shield and waste package impacted by an igneous event is expected to fail such that neither can continue to function as a barrier to seepage or radionuclide transport (SNL 2007 [DIRS 177430] Section 6.4.8.3.5).

Magma velocities must exceed 15 to 20 meters per second (SNL 2007 [DIRS 177430], Section 6.4.8.3.3) to cause significant movement of waste packages. Magma velocities are estimated to be in the range of 1 to 5 meters per second (SNL 2007 [DIRS 177430], Section 6.4.8.3.2).

4.1.5 Waste Form Misload Probability

The TAD canisters will be loaded with commercial SNF at the respective power facilities according to loading curves developed for the particular waste form and shipped to the repository for aging and/or disposal. Uncanistered fuel assemblies will be loaded at the repository using a TAD canister similar to the site-specific canister design. Loading curves, which are functions of burnup and enrichment, are the loci of values delineating the region of acceptable burnup/enrichment combinations for postclosure criticality control of the TAD canisters. In order to accommodate multiple canister criticality control design configurations, focus was placed on the TAD canister specification requirements in the reference cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for Transportation Aging and Disposal Canister and Related Waste Package Physical Attributes Basis for Performance Assessment* (SNL 2007 [DIRS 179394], Section 4.1.1.5) in developing a robust loading curve that would bound all of the potential design variants. In applying this methodology, the loading curve is generated once and the assigned burnup values of all assemblies considered for loading into the TAD canister are compared directly against this loading curve. Assemblies having burnup values in the *unacceptable* range must be loaded into waste packages or canisters with additional reactivity control mechanisms (e.g., disposal control rod assemblies). The process for developing the criticality loading curves for each canister configuration and range of commercial SNF characteristics is documented in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008 [DIRS 182788]). The recommended loading curves use a conservative design basis configuration that accounts for neutron absorber plate degradation (thickness subjected to 10,000 years of corrosion), geometric rearrangement conducive to criticality, 75% credit for the neutron absorber material distributed throughout the absorber plates, all fuel represented as the most reactive assembly type and the worst (highest criticality potential) assembly design[4]. Therefore, these analyses demonstrate that an intact, fully flooded with water (i.e., a neutron moderator) TAD canister waste package configuration fabricated and loaded as designed will not achieve criticality (SNL 2008 [DIRS 182788], Section 6.2.2).

Waste form misloads for commercial SNF occur when assemblies with initial enrichment and burnup parameters in the restricted (*unacceptable*) loading curve range are improperly loaded into waste packages not designed for these particular parameters. This type of loading curve violation is represented in this analysis as an assembly misload error. An analysis of commercial SNF assembly misload probabilities was documented in *Commercial Spent Nuclear Fuels Waste Package Misload Analysis* (BSC 2003 [DIRS 166316]). Results from this analysis assign the probability of misloading an SNF assembly into a 21-PWR Absorber Plate Waste Package as 1.18×10^{-5} (BSC 2003 [DIRS 166316], Table 41). However, neighboring assemblies that have low reactivity values may provide partial compensation for the excess reactivity from the incorrectly loaded assembly. Given that a misload occurs, the likelihood of the misloaded configuration having potential for criticality has been shown to be 0.014 from results of a sensitivity calculation of that potential (SNL 2008 [DIRS 182788], Section 7). The cited analysis is used as a surrogate for misloading waste forms in a TAD canister as the misloading of an assembly into a TAD canister requires a similar improper selection of an assembly with characteristics (burnup and enrichment) in the unacceptable range of the loading curve.

The probability of misloading assemblies in the 44-BWR TAD canister is insignificant as the entire BWR component of the license application waste stream (CRWMS M&O 2000 [DIRS 138239], Section 6) is in the acceptable region of the loading curve map (SNL 2008 [DIRS 182788], Section 6.1.1.1.3). The shape of the defense high-level waste (DHLW) glass canisters differs significantly from the DOE-owned SNF canisters and the various DOE waste forms also differ significantly in size and shape (Radulescu et al. 2004 [DIRS 165482], Sections 2 and 3; DOE 2007 [DIRS 183392], Section 5.2). Thus, the waste forms and canisters can be readily distinguished by visual inspection and misloading of waste forms in DOE-owned SNF canisters or misloading the canisters into codisposal waste packages is considered very improbable. Therefore, the waste form misload probability for codisposal waste packages is considered insignificant.

4.1.6 Characteristics of the Drift, Waste Package, and Drip Shield

The emplacement drift has a nominal diameter of 5.5 meters (5,500 mm) (SNL 2007 [DIRS 179466], Table 4-1, Item 01-10). Within the drift, the steel support beams and associated ballast form a level invert with a surface height of 52 inches (1,320.8 mm) above the lowest part of the drift (SNL 2007 [DIRS 179354], Figure 4-1). The drip shield is a free-standing titanium structure made of top and side plates supported by a framework that sits on the invert (SNL 2007 [DIRS 179354], Section 4.1.2). The initial plate thickness is 15 mm (SNL 2007 [DIRS 179354],

---

[4] Variations in fuel assembly lattice design were evaluated in *Evaluation of Neutron Absorber Materials Used for Criticality Control in Waste Packages* (BSC 2006 [DIRS 180664]). This evaluation was performed to assess which fuel assembly lattice design would result in the highest k_{eff} values when loaded in a waste package configuration. Fuel assembly lattices were varied using Babcock & Wilcox, Westinghouse, and Combustion Engineering geometric arrangements in pure water. The results of this evaluation indicate that the Babcock & Wilcox 15×15 fuel assembly design is one of the most reactive designs (BSC 2006 [DIRS 180664], Section 7); therefore, it is selected as the representative bounding PWR assembly design.

Table 4-2, Item 07-04). The drip shield has an external height for the overlap section of 113.62 inches (2,886 mm) (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), rounded up to 2,890 mm. The internal height of the drip shield, defined as the distance from the invert floor to the lowest point on the underside of the top of the drip shield, is 106.93 inches (2,716 mm) (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), which is rounded to 107 inches (2,717.8 mm). The clearance between the crown (top) of the drip shield and the drift roof is 50.37 inches (1,279 mm) (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), rounded to the nearest inch or 1,270 mm. The nominal waste package separation in the drift is 10 cm (100 mm) (SNL 2007 [DIRS 179354], Table 4-4, Item 05-02). The waste packages consist, among other components, of an outer corrosion barrier in the form of a cylindrical shell of Alloy 22 and an inner vessel of stainless steel (SNL 2007 [DIRS 179394], Table 4-3). Nominal dimensions for the outer corrosion barrier of the various waste package variants are listed in Table 4.1-3. The nominal lengths of waste packages in Table 4.1-3 include the OCB lifting feature but extra length has minimal impact on the calculations for seismic faulting effects. The outer corrosion barrier vessel and lid thickness for all DOE-owned SNF waste package variants is ≥ 25 mm (SNL 2007 [DIRS 179394], Table 4-3; SNL 2007 [DIRS 179567], Tables 4-8 through 4-10). Thus, the fuel matrix in the TAD canister and surrogate site-specific canister/basket are expected to be very similar to the fuel matrix in the former 21-PWR Absorber Plate and 44-BWR Absorber Plate design variants, respectively (SNL 2007 [DIRS 179394], Sections 1 and 4.1.1.2).

Seismic faulting can generate a large number of possible dynamic response scenarios in a drift. A reasonable approach (rationale in Section 6.4.1, Footnote 5) for simplifying the analyses was to calculate clearances excluding the pallet elevations (SNL 2007 [DIRS 176828], Section 6.11.1.1). The combined clearance between the crown of the drip shield and the roof of the drift (1,270 mm), and between the top of the waste package and the bottom of the drip shield (shown in Table 4.1-3) determines the maximum fault displacement that could occur before the waste packages are potentially damaged or breached through a shearing mechanism. This analysis selected the smaller of these clearances since drift collapse is likely in the lithophysal zone during a disruptive event associated with fault displacement.

Table 4.1-3. Maximum Allowable Displacement with Drift Collapse for an Intact Drip Shield

Package Type	Outside Diameter of OCB (mm)	Nominal Length (mm)	Clearance Without Pallet (mm)
Commercial SNF TAD Canister	1,881.6	5,850.1	836
Codisposal Short	2,044.7	3,697.4	673
Codisposal Long	2,044.7	5,303.9	673
Codisposal-MCO	1,749.3	5,278.6	969

Sources: SNL 2007 [DIRS 179394], Table 4-3, for outside diameter of OCB and for nominal length of the TAD waste package; SNL 2007 [DIRS 179567], Tables 4-8 through 4-10, for the outside diameter of OCB and nominal length of the codisposal waste package types.

NOTES: Clearance without the pallet is calculated as the interior height of the drip shield (2,717.8 mm) minus the outside diameter of the waste package OCB, rounded to three significant digits (listed in Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: “Tables by WP Type,” rows 14 to 24).

MCO = multicanister overpack.

4.1.7 Waste Package and Drip Shield Interactions with Seismic Events

The number of waste packages that could be emplaced on faults in the repository is evaluated in Section 6.4.1 following the analysis method from DTN: MO0705FAULTABS.000 [DIRS 183150], file: *Fault Displacement Abstraction for Criticality.xls*, spreadsheet: “Tables by WP Type” adjusted for the inventory from Table 4.1-2 and dimensions from Table 4.1-3.

The probability of drip shield damage or failure from seismically induced rockfall is developed through fragility curves for the drip shield plates and framework that is documented in DTN: MO0703PASDSTAT.001 [DIRS 183148], file: *Plate Fragility Analysis.xls*, spreadsheet: “Summary,” and file: *Frame Fragility Analysis.xls*, spreadsheet: “Summary,” respectively. Significant failure probabilities were developed for the nondegraded drip shields subjected to 100% rockfall loads at exceedance frequencies of approximately 10^{-8} per year based on the bounded hazard curves. Likewise, with the exception of very large blocks, rockfall in nonlithophysal zones does not cause waste package damage (SNL 2007 [DIRS 179476], FEP 1.2.03.02.0B).

4.1.8 Emplacement Drift Information

Emplacement drift information is required to properly assign seismic information to the two geologic zones – lithophysal and nonlithophysal. The lithophysal and nonlithophysal fractional areas are calculated by dividing the emplacement drift area of both geological zones by the total drift area. The drift emplacement area by geological unit is given in Table 1 of the reference cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for Subsurface Facilities* (SNL 2007 [DIRS 179466], Table 4-1, Item 01-01). This information is summarized in Table 4.1-4.

Table 4.1-4. Drift Emplacement Area by Geological Unit

Geological Unit	Drift Emplacement Area (square meters)	Percentage of Area
1. Tptpul (lithophysal)	224,398	4.5
2. Tptpmn (nonlithophysal)	616,003	12.4
3. Tptpll (lithophysal)	4,013,268	80.5
4. Tptpln (nonlithophysal)	129,483	2.6
5. Total Lithophysal Sum of rows 1 and 3	4,237,666	85.0
6. Total Nonlithophysal Sum of rows 2 and 4	745,486	15.0
7. TOTAL Sum of rows 5 and 6	4,983,152	100.0

Source: Table 1 of the reference cited in SNL 2007 [DIRS 179466], Table 4-1, Item 01-01.

4.1.9 Seismic PGV Values and Exceedance Frequencies

The intensity of a seismic event is defined in terms of the peak ground velocity of the first horizontal component of the ground motion, denoted as PGV-H1 or more simply as PGV (SNL 2007 [DIRS 176828], Section 4.1). The potential for Engineered Barrier System damage from seismic events is normally correlated with the PGV value (e.g., Table 6-78 from *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828])). Two correlations of PGV values versus seismic exceedance frequency are used in the screening analysis (i.e., values derived from a bounded hazard curve (DTN: MO0501BPVELEMP.001 [DIRS 172682]) and an unbounded hazard curve (SNL 2007 [DIRS 176828], Section 6.4.3)) where "bounded" and "unbounded" are discussed in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6.4.3) and abstracted as follows:

> …PGV values in excess of 5 m/sec are extremely large and may not be physically realizable for the seismic sources and geologic conditions in and around Yucca Mountain. In particular, the physical properties of the lithophysal rocks at the emplacement drift level are expected to provide physical limits on the PGV values at the repository location. A reevaluation using maximum shear strains to define a distribution of horizontal PGV values that are consistent with observations resulted in a bounded (more realistic) hazard curve for horizontal PGV values at the repository waste emplacement level.

The two sets of values are shown in Table 4.1-5 and in Figure 4.1-1 and given in Output DTN: MO0705CRITPROB.000, file: *PGV vs Exceedance Freq.xls*, spreadsheet: "Data Source."

4.1.10 Yield Factor for Radiolytic Species Generation

The radiolytic yield of any given chemical species is characterized by a single parameter, "G," that represents the number of molecules of a chemical species produced per 100 eV of absorbed radiation energy in the volume containing the irradiated environment. The average value for the net H_2 generation rate, $G_\gamma(H_2)$, in a moist He volume simulating a TAD canister waste package with residual water was estimated as 0.49 based on values from Green (1994 [DIRS 181678], Table 4-2).

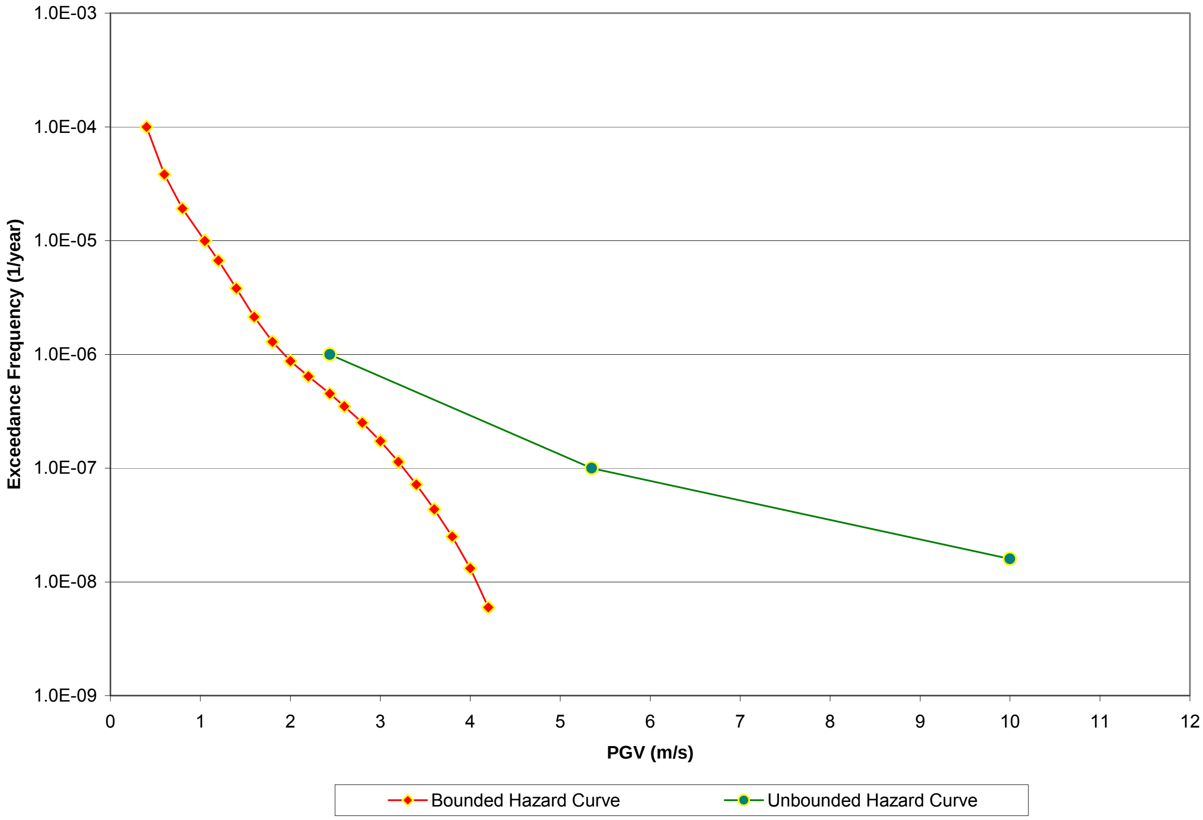
4.1.11 Radiolytic Dose in TAD Canisters

The gamma dose for the radiolysis analysis was derived from the source strength of commercial SNF assemblies in a 21-PWR waste package given in Table 6 of the reference cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for Transportation Aging and Disposal Canister and Related Waste Package Physical Attributes Basis for Performance Assessment Requirements Analysis for Subsurface Facilities* (SNL 2007 [DIRS 179394], Table 4-1, Item 03-10). The gamma dose and resultant energy deposition rate from commercial SNF is two or more orders of magnitude larger than the neutron dose as noted in the cited Table 6.

Table 4.1-5. PGV Values versus Seismic Exceedance Frequency

Bounded Hazard Curve [a]		Unbounded Hazard Curve [b]	
PGV (m/s)	Exceedance Frequency (1/yr)	PGV (m/s)	Exceedance Frequency (1/yr)
.4019	1.00×10^{-4}	2.44	1.00×10^{-6}
0.6	3.826×10^{-5}	5.35	1.00×10^{-7}
0.8	1.919×10^{-5}	10.0	1.597×10^{-8}
1.05	9.955×10^{-6}		
1.2	6.682×10^{-6}		
1.4	3.812×10^{-6}		
1.6	2.136×10^{-6}		
1.8	1.288×10^{-6}		
2.0	8.755×10^{-7}		
2.2	6.399×10^{-7}		
2.44	4.518×10^{-7}		
2.6	3.504×10^{-7}		
2.8	2.507×10^{-7}		
3.0	1.731×10^{-7}		
3.2	1.137×10^{-7}		
3.4	7.168×10^{-8}		
3.6	4.362×10^{-8}		
3.8	2.508×10^{-8}		
4.0	1.319×10^{-8}		
4.20	5.967×10^{-9}		

Sources: [a] DTN: MO0501BPVELEMP.001 [DIRS 172682], file: *Bounded Horizontal Peak Ground Velocity Hazard at the Repository Waste Emplacement Level.xls*, spreadsheet: "Bounded Horizontal PGV Hazard."
[b] Unbounded Hazard Curve: SNL 2007 [DIRS 176828], Section 6.4.3.



Sources: DTNs: MO0501BPVELEMP.001 [DIRS 172682]; file: *Bounded Horizontal Peak Ground Velocity Hazard at the Repository Waste Emplacement Level.xls*, spreadsheet: "Bounded Horizontal PGV Hazard;" Unbounded Hazard Curve: SNL 2007 [DIRS 176828], Section 6.4.3.

Figure 4.1-1. Seismic Exceedance Frequency versus PGV Value

4.1.12 Hydrogen Deflagration

Hydrogen concentrations of 4% or greater by volume are required for deflagration in air (Coward et al. 1952 [DIRS 182138], Figure 7, Table 3; Kuo 1986 [DIRS 170633], Table 4.5). The minimum oxygen concentration capable of supporting a flame front is approximately 4 vol % (Coward et al. 1952 [DIRS 182138], Table 44). Gas temperatures resulting from a hydrogen deflagration are approximately 350°C (Coward et al. 1952 [DIRS 182138], p. 15). Gas pressure ratios resulting from a hydrogen deflagration are approximately 1 to 4 times the initial pressure (Coward et al. 1952 [DIRS 182138], p. 12).

4.1.13 Vibratory Seismic Ground Motion

Stress corrosion cracking from high residual (tensile) stress threshold (RST) is expected to be the failure mode of waste packages subjected to impact processes due to vibratory induced ground motion events (SNL 2007 [DIRS 178851], Section 8.2). The damaged or deformed area that exceeds an RST value is conceptualized to result in a tightly spaced network of stress corrosion cracks. The residual tensile stress threshold is often shortened to the residual stress threshold or RST, with the understanding that the principal residual stress must always be tensile to initiate SCC. The results from each calculation of structural response of the waste package seismic events are calculated for three discrete values of the RST level for Alloy 22 (i.e., 90%, 100%, and 105% of the yield strength of Alloy 22). The intermediate value of 100% has been included because damaged areas may be a nonlinear function of RST. For convenience, these three values are referred to as the 90%, 100%, and 105% RST, respectively (SNL 2007 [DIRS 176828], Section 6.1.4). The probabilities evaluated at the 90% RST level give a conservative estimate and are used for the screening evaluations.

The estimate for the probability of damage to the OCB of a TAD canister waste package due to seismic vibratory induced ground motion is given as 0.118 (SNL 2007 [DIRS 176828], Section 6.5.1.2, DTN: MO0703PASDSTAT.001 [DIRS 183148], file: *Kinematic Damage Abstraction 23-mm Intact.xls*, spreadsheet: "Probability of Damage") at an RST value set at 90% of the yield strength of Alloy 22 for nondegraded waste package components. The probability of damage to the OCB of a TAD canister waste package is zero at an RST value set at 100% of the Alloy 22 yield strength.

The probability of damage to the OCB of a codisposal waste package due to seismic-induced vibratory ground motion is discussed in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6.6.1.2). From the kinematic response of the codisposal waste packages to vibratory induced ground motions, the probability of damage to the OCB is nonzero over a range of PGV levels for the three RST levels as given in Table 4.1-6. The codisposal waste package will remain undamaged for zero probability states.

Table 4.1-6. Probability of Damage for Intact Codisposal Waste Package

PGV Level (m/s)	Residual Stress Threshold as Percentage of Yield Strength		
	90%	100%	105%
0.364	0	0	0
0.4	0.029	0	0
1.05	0.559	0	0
2.44	0.941	0.147	0
4.07	1	0.412	0

Source: DTN: MO0703PASDSTAT.001 [DIRS 183148], file: *CDSP Kinematic Damage Abstraction 23-mm Intact.xls*, spreadsheet: "Probability of Damage - New."

Significant rockfall onto and around the drip shields is likely to occur during seismic vibratory events, which have the potential of rupturing the drip shields. The likelihood of such damage in the lithophysal zones is discussed in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6.8.2.2). The probability of commercial SNF and codisposal waste packages breaching from the combination of drip shield rupture and localized corrosion, considering the environmental conditions for localized corrosion of the waste package OCB (see Section 4.1.14), is developed from DTN: MO0712PANLNNWP.000 [DIRS 184480], file: *mo0712panlnnwp 000.zip*, folders: 3D and 4D. The conditional distribution of probability that waste packages will undergo localized corrosion is obtained from an intermediate product of TSPA. Specifically, these results consist of sets of simulated outcomes for 300 realizations over a set of dominant epistemic parameters, with drip shields removed, in which the responses for a group of waste packages (i.e., localized corrosion or not) are calculated for every realization. Five sets of 300 outcomes are used, corresponding to the five percolation "bins" used in TSPA to represent variability and uncertainty in percolation flux (SNL 2008 [DIRS 184433], Section 6.2.12.1[a]). The computational method for evaluation of the combined drip shield rupture and waste package OCB localized corrosion probability is documented in File: *Localized_Corrossion.zip* (files: *CSNF_bin[x].txt* for x = 1, 2, 3, 4, and 5; *CDSP_bin[x].tx*t for x = 1, 2, 3, 4, and 5) (DTN: MO0709TSPALOCO.000 [DIRS 182994]; SNL 2007 [DIRS 183478], Appendix O). The calculation uses input from the host-rock lithology, localized corrosion probability, waste package temperature and relative humidity, temperature effect from drift collapse, and uncertainty in the parameters that describe seepage chemistry. Note that these intermediate results are available only in separate sets for the lithophysal and nonlithophysal units, so to obtain the total probability distribution across the entire repository the analysis must be done for each set with the method and results documented in (SNL 2007 [DIRS 184078], Appendix B).

Analyses of large, single-block (28.29 metric tons) impacts of rocks in the nonlithophysal zone show that such impacts may cause the drip shield to buckle and potentially contact the waste package outer corrosion barrier. The analysis indicated that waste package damage could occur for the most severe events at a PGV level of 5.35 m/s (SNL 2007 [DIRS 178851], Table 6-153).

4.1.14 Physical Properties

Properties of various materials that may be used in the fabrication of waste packages and canisters are documented in this section. Physical properties of gases used in this analysis are listed in Table 4.1-7. The waste package outer barrier is constructed of Alloy 22, which has a corrosion rate that ranges from 0 to 15 nm/yr (SNL 2007 [DIRS 178519], Figure 6-10) and the probability of general corrosion occurring is low (SNL 2007 [DIRS 178519], Section 8.1). Stress corrosion cracks can develop in Alloy 22, however, and propagate at a rate of 1.1×10^{-9} mm per second (SNL 2007 [DIRS 181953], Table 6-6, Section 8.1.2). This rate is essentially independent of the stress intensity factor (SNL 2007 [DIRS 181953], Figure 6-9 and Section 8.1.2).

Localized corrosion in the form of pitting and crevice corrosion can occur on exposed surfaces of the waste package OCB provided an appropriate aqueous environment is present (SNL 2007 [DIRS 179476], FEP 2.1.03.03.0A). Seepage water through ruptured drip shields can provide the basis for such an environment to develop. Once localized corrosion occurs, it propagates at a median rate of 127 µm per year with a lowest percentile of 12.7 µm per year (DTN: MO0703PAGENCOR.001 [DIRS 182029], file: *LC_Propagation.pdf*, Table 1). Thus localized corrosion, once initiated, can penetrate the waste package OCB in less than 1000 years.

The absorber material designated for the TAD canisters is borated stainless steel (Orrell 2007 [DIRS 182643]) produced by powder metallurgy that results in a near-optimal dispersion of boron throughout the material (ASTM A 887-89 Grade A [DIRS 178058], pp. 1 to 4). Corrosion rates for neutron absorber materials measured in aqueous environments simulating expected repository environments are listed in Table 4.1-8. The initial thickness specified for the borated stainless steel absorber plates in the TAD canisters is 11 mm (SNL 2007 [DIRS 179394], Table 4-1, Item 03-10). Based on the average corrosion rate of 0.0271 µm/yr (set with exposure time > 100 hr showing the highest average) from Table 4.1-8, the absorber plate thickness after 10,000 years would be approximately 10 mm.

Table 4.1-7. Physical Properties of Gases

Properties of Gases		
Property	**Value**	**Source**
Standard atmosphere pressure	0.101 MPa	Parrington 1996 [DIRS 103896], Physical Constants
Molecular weight of water	18.015×10^{-3} kg/mole	Parrington 1996 [DIRS 103896], Physical Constants
Molecular weight of helium	4.003×10^{-3} kg/mole	Parrington 1996 [DIRS 103896], Physical Constants
Density of water vapor	Function of temperature (kg/m^3)	ASME 1993 [DIRS 108050], Table 1
Density of helium	Function of temperature (kg/m^3)	Holman 1997 [DIRS 101978], Table A-6

Table 4.1-8. Corrosion Rates of Waste Package Materials

Absorber Material	Corrosion Rate		Notes
Ni–Gd Alloy [a]	Average value	0.056 μm/yr	30°C, J-13 solutions
Ni–Gd Alloy	Average value	0.307 μm/yr	60°C, J-13 solutions
Borated Stainless Steel – 400 hour test [b] ASTM 304B4 Grade A alloy [c]	Average value	0.0271 μm/yr	Range 25°C to 90°C
Stainless Steel Type 316L [d]	Median value	0.003 μm/yr	30°C fresh water

Sources: [a] DOE 2004 [DIRS 168434], Table 17.
[b] DTN: MO0706ECTBSSAR.000 [DIRS 181380], Table 5.
[c] Orrell 2007 [DIRS 182643].
[d] DTN: MO0409SPAACRWP.000 [DIRS 172059], file: *aqueous-316L.xls*, spreadsheet: "freshwater."

4.1.15 Criticality Potential of Waste Forms

As discussed in Section 1.4, a configuration class (a set of similar configurations whose composition and geometry are defined by specific parameters) is considered to have *potential for criticality* if the probability of the configuration class formation is above a specified probability screening criterion. For configurations with potential for criticality (i.e., probability of the configuration occurring is above the screening criterion), an additional evaluation of the range of configuration class parameters may be necessary to determine if the maximum effective neutron multiplication factor (k_{eff}) range could exceed the critical limit for the waste form. If such is the case, additional control measures may be required to assure that the maximum k_{eff} range is below the critical limit for the waste form.

The criticality potential of waste forms in the external near-field location depends on whether the fissile mass that can be accumulated in this location over 10,000 years after closure can exceed the minimum critical mass for the waste form in that environment. Masses accumulated in the external far-field cannot exceed the near-field quantities since the near-field mass is the source for any far-field accumulation. Fissile mass accumulations from diffusive releases into the invert have been evaluated and documented in DTN: MO0604SPANOMIN.000 [DIRS 182944], file: *CSNF Results.xls*, spreadsheet: "Table for Report" and file: *DOE SNF Results.xls*, spreadsheet: "Table for Report" with the median values shown in Table 4.1-9 for the early engineering barrier failure event scenario. The fissile mass accumulations in the invert or host rock for the igneous and seismic scenarios in Table 4.1-9 were obtained from DTNs: MO0609SPAINOUT.002 [DIRS 179645], file: *Dissolved U and Pu acc total with sensitivities.xls*, and MO0705PHREEMOD.000 [DIRS 183622], file: *Mass Accumulated.xls*.

Table 4.1-9. Fissile Mass Accumulation in Invert or Host Rock

Waste Form	Plutonium Accumulation (kg)	Total Uranium Accumulation (kg)	^{235}U Accumulation (kg)
Nominal (Early Failure) Scenario [a]			
Commercial SNF	1.49×10^{-7}	2.68×10^{0}	3.98×10^{-2}
DOE3 (N Reactor)	NA	3.32×10^{2}	5.32×10^{0}
DOE1 (FFTF)	3.57×10^{-4}	1.03×10^{1}	8.31×10^{-2}
DOE9 (TMI II)	NA	1.76×10^{1}	4.92×10^{-1}
Seismic Scenario			
DOE1 (FFTF) [b]	8.63×10^{-4}	6.50×10^{1}	2.74×10^{-1}
Igneous Scenario			
Commercial SNF [b]	7.31×10^{-7}	7.48×10^{1}	9.72×10^{-1}
DOE3 (N Reactor) [c]	NA	1.09×10^{-1}	1.37×10^{-3}
DOE1 (FFTF) [c]	6.34×10^{-3}	1.72×10^{1}	1.12×10^{0}
DOE9 (TMI II) [c]	NA	3.07×10^{1}	8.08×10^{-1}

Sources: [a] DTN: MO0604SPANOMIN.000 [DIRS 182944], file: *CSNF Results.xls*, spreadsheet: "Table for Report," file: *DOE SNF Results.xls*, spreadsheet: "Table for Report."
[b] DTN: MO0705PHREEMOD.000 [DIRS 183622], file: *Mass Accumulated.xls*.
[c] DTN: MO0609SPAINOUT.002 [DIRS 179645], file: *Dissolved U and Pu acc total with sensitivities.xls*.

FFTF = Fast Flux Test Facility; NA = not available.

The minimum mass necessary for external criticality (based on a critical limit of 0.96) for all the cases evaluated is summarized in Table 4.1-10, which shows that the masses in the external environment necessary for criticality are higher, often much higher, than the predicted fissile mass accumulation that is also shown in Table 4.1-10. The largest percentage (71%) of a minimum critical mass accumulated in the invert or host rock was from commercial SNF for a seismic scenario. The minimum critical masses were calculated using optimal conditions for criticality that are improbable in the invert or host rock.

The DOE-owned SNF waste forms addressed in *Geochemistry Model Validation Report: External Accumulation Model* (SNL 2007 [DIRS 181395]) (i.e., N Reactor (DOE3), TMI-II (DOE9), and Fast Flux Test Facility (FFTF) (DOE1)) make up approximately 90% of the metric tons of heavy metal in the DOE-owned SNF inventory expected to be stored in the repository. Some of the other DOE-owned SNF with high enrichments, such as Shippingport light water breeder reactor (DOE5) and Ft. St. Vrain (DOE6), are also not expected to be a concern due to the corrosion resistance of the waste form (SNL 2007 [DIRS 181395], Section 6.9.3[a]).

Table 4.1-10. Summary of External Criticality Results - Minimum Mass for a Critical Limit of 0.96

Scenario	Waste Package Type	Calculated Accumulation or Mass Released from Waste Package	Mass of Uranium or Plutonium (for FFTF) Required to Achieve k_{eff} = 0.96			
		Uranium or Plutonium mass (kg)	**Invert (kg)**	**Fractured Tuff (kg)**	**Lithophysae Array (kg)**	**Large Lithophysae (kg)**
Seismic	N-Reactor	Not calc [a]	266,000	∞ [b]	Not calc	Not calc
	TMI-II Fuel	Not calc	350	∞	Not calc	Not calc
	Commercial SNF	90.3 [c]	126	∞	Not calc	Not calc
	FFTF (Plutonium mass)	0	1.66	4.3	Not calc	Not calc
Igneous	N Reactor	0.109	∞	∞	∞	∞
	TMI-II	30.7	538	∞	∞	∞
	Commercial SNF	74.8	159	∞	1390	∞
	FFTF (Plutonium mass)	6.34×10^{-3}	1.66	4.3	4.0	2.2

[a] "Not calc" means that this scenario was of little interest given that it was bounded by another scenario. In most cases,
this simply meant that, if commercial SNF waste was very subcritical, then TMI-II and N Reactor had to be subcritical also.

[b] "∞" means that an infinite amount of fissile waste released in this model will not produce an arrangement that can reach a k_{eff} of 0.96.

[c] Maximum mass released from the waste package.

Source: The mass required to achieve k_{eff} = 0.96 is from DTN: MO0705SCALEGEO.000 [DIRS 183634], files: *DOEF.xls*, "SJN_All" and "UBN_All;" file: *FFTF.xls*; "PBL," "PBN," "PJF," "PBV;" file: *CSNF.xls*, "BBN5," "UBL5." Table values were abstracted from (SNL 2007 [DIRS 181395], Table 6.9-1 [a]).

Fort St. Vrain SNF (DOE6) SNF fuel particles have an integral silicon carbide (SiC) protective layer that not only retains the fission products but also protects the uranium and thorium dicarbide (ThC_2) from oxidation and hydrolysis (DOE 2003 [DIRS 166027], p. 48). Comparative analyses have indicated that the Fort St. Vrain fuel has the lowest degradation rate of all DOE-owned SNF and should behave significantly better in terms of fissile material dissolution, transport, and accumulation. Graphite moderated reactors such as Fort St. Vrain are considered as "converter" reactors where fissile isotopes are produced (^{233}U for this system) as the ^{235}U is consumed (Radulescu et al. 2004 [DIRS 165482], Section 3.2.8.2). A canister loaded with five Fort St. Vrain blocks contains sufficient quantities of ^{233}U to have criticality potential in solution; however, a mechanism to separate the uranium from within the SiC coated fertile particles, and then a mechanism to accumulate it in a concentrated fissile mass in a favorable geometry is not credible.

A number of studies have indicated both air and water oxidation of uranium and thorium oxide fuel pellets [$(Th, U)O_2$] such as used in the Shippingport light water breeder reactor fuel (DOE5 proceeds more slowly than for pure uranium oxide (UO_2), and decreases with decreasing UO_2 content in the $(Th, U)O_2$ (DOE 2003 [DIRS 166027], p. 33). Tests have shown that the thorium oxide pellets in the Shippingport light water breeder reactor fuel have an excellent resistance to mobility with an estimated solubility of 10^{-14} mol/L at 25°C and pH > 5 (DOE 2003 [DIRS 166027], p. 32). With the less aggressive degradation rate, a mechanism to separate the uranium, transporting, and accumulation in a favorable geometry is also not credible.

DOE fuel groups in Table 4.1-2 (DOE2, DOE4, DOE7, and DOE8) representing UZrHx (TRIGA), high enriched uranium oxide (Shippingport PWR), aluminum-based (advanced test reactor), and U-Zr/U-Mo alloy (Fermi), respectively, have not been analyzed in detail for external fissile mass transport and accumulation. However, these waste forms are not expected to result in an increase in the total probability of criticality in the external environment based on:

- Consideration of the processes that must occur to allow advective seepage into a DOE-owned SNF canister without substantial drainage to allow degradation of the internal components and waste form
- Use of conservative modeling parameters to facilitate fissile material transport to the external environment
- Use of bounding modeling parameters to maximize the criticality potential.

Some of the conservatisms that are part of the basis of this conclusion are as follows:

- The material degradation and release model (SNL 2007 [DIRS 181165]) uses constant corrosion rates for the SNF, however laboratory experiments on the surface structure of commercial SNF during dissolution have shown that UO_2 dissolution is accompanied by the formation of a protective layer of secondary phases that retards further corrosion (SNL 2007 [DIRS 181165], Section 6.6.2). Therefore, the release of uranium from the fuel would be slower and therefore less would accumulate in the external environment.
- Experimental and field data indicate that actinides would adsorb on or incorporate into alteration products that form in the waste package (SNL 2007 [DIRS 181165], Section 6.6.3). This solid solution formation and adsorption would tend to lower actinide concentrations below those predicted by the equilibrium geochemistry evaluations and would delay release from the waste package.

In addition, the reaction rate of U-Mo is at least an order of magnitude lower than for uranium metal (DOE 2000 [DIRS 152658], Figure 5-2), which is considered as the bounding waste form for corrosion of metal fuels and alloys (DOE 2000 [DIRS 152658], Section 5). (One caveat concerning the corrosion rate of U-Mo alloy is that, for very long times, the alloy may be subject to discontinuous failure phenomena where the corrosion rate increases substantially but should not affect the retention characteristics.) The two remaining DOE-owned SNF waste forms, TRIGA (DOE2) and Shippingport PWR (DOE4) consist of Zirconium-based metallic mixtures. The TRIGA SNF consists of a metallic uranium dispersed in a zirconium-hydride matrix (Radulescu et al. 2004 [DIRS 165482], Section 3.2.2). The Shippingport PWR SNF consists of a uranium-zirconium alloy sandwiched between Zircaloy plates (Radulescu et al. 2004 [DIRS 165482], Section 3.2.3). The corrosion rate of Zircaloy is sufficiently low to be considered as insoluble for degradation studies (BSC 2004 [DIRS 169982], Section 6.2.5) minimizing the degradation rate for these latter two waste forms and, thus, the quantity of material available for accumulation in the external environment. An assumption in the material degradation and release model (SNL 2007 [DIRS 181165]) is that the cladding and DOE-owned

SNF canister fail immediately, whereas a more likely scenario is that the failure would take place over many years. This would also delay the release of actinides.

Likewise, many conservative assumptions are used to simplify the critical mass calculations presented in Table 4.1-10. For the analysis of commercial SNF and low-enriched DOE fuels discussed in *Geochemistry Model Validation Report: External Accumulation Model* (SNL 2007 [DIRS 181395]), the conservatisms are appropriate, because the results show that a criticality is unlikely. However, for the higher enriched DOE fuels, a more realistic modeling of the criticality potential will most likely be required to generate conservative but realistic results. Thus, it is concluded that the likelihood of achieving a configuration in the external environment with potential for criticality very low.

Degradation analyses indicate that, due to the boron in borated stainless steel having a very low solubility within the iron matrix of the steel, the boron is present as separate chromium boride particles instead of a solid solution (SNL 2007 [DIRS 181165], Section 6.3.3). This type of material does not dissolve into the aqueous solution during degradation of the steel but is left behind as insoluble products during corrosion.

For selected DOE-owned SNF waste forms, the neutron absorber material is in a distributed form in the waste form canister as the absorber material is added in the form of shot at the time of waste form loading per Section 5.2.3 of *Criticality Potential of Intact DOE SNF Canisters in a Misloaded Dry Waste Package* cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for DOE SNF/HLW and Naval SNF Waste Package Physical Attributes Basis for Performance Assessment* (SNL 2007 [DIRS 179567], Table 4-1, Item 03-02).

Representative cases for the degradation and reconfiguration of the internal structures and waste forms in a waste package have been addressed in numerous analyses for the various SNF waste forms. These results are summarized in the following documents: *CSNF Loading Curve Sensitivity Analysis* (SNL 2008 [DIRS 182788]); *DOE SNF Phase I and II Summary Report* (BSC 2004 [DIRS 165482]); *Intact and Degraded Mode Criticality Calculations for the Codisposal of TMI-2 Spent Nuclear Fuel in a Waste Package* (BSC 2004 [DIRS 168935]); *Intact and Degraded Mode Criticality Calculations for the Codisposal of ATR Spent Nuclear Fuel in a Waste Package* (BSC 2004 [DIRS 171926]); and *Criticality Analyses for FFTF Fuel with Advanced Neutron Absorber Material* (SNL 2007 [DIRS 183927], Section 6.9).

The DOE-owned SNF waste forms that require plate type neutron absorber materials ((Ni-Gd), ASTM B 932-04 [DIRS 168403], pp. 1 to 2) are mixed oxide (DOE1), $UZrH_x$ (DOE2), U/Th Oxide (DOE5), aluminum-based DOE-owned SNF (DOE7), and U-Zr/U-Mo (DOE 8) (DOE 2004 [DIRS 170071], Section 2.1.11). The absorber material for the DOE5 and DOE8 SNF waste forms consists of a combination of both plates and shot and, thus, the absorber misload probability is considered insignificant. Thus, the DOE1, DOE2, and DOE7 waste forms are the only ones for which configurations with criticality potential have a non-trivial probability of absorber misload. These three latter waste forms have the possibility of exceeding the critical limit of the respective waste forms for degraded configurations (BSC 2004 [DIRS 171926], Section 6.2.1; Radulescu et al. 2004 [DIRS 165482], Sections 10.1.2.3 and 10.2). However, the analysis shows that the addition of gadolinium-doped metal shot to the canister will provide adequate criticality control for degraded configurations. Thus, the criticality potential of aluminum-based DOE-owned SNF (DOE7), mixed oxide DOE-owned SNF (DOE1), and uranium-zirconium hydride (DOE2) configurations would be insignificant on condition that gadolinium shot is added to the canisters for these waste forms. Further criticality analysis (SNL 2007 [DIRS 183927], Section 6.3.9) of the DOE1 waste form (FFTF SNF) has indicated the necessity of having a distributed neutron absorber in the disposal canister that would allow the probability of an absorber misload for the DOE1 waste form to be considered as insignificant.

The gadolinium in the DOE-owned SNF canisters forms phosphate or carbonate corrosion products (SNL 2007 [DIRS 181165], Section 6.3.16), both of which have low solubilities. Essentially all of the absorber material is retained in the waste package for the DOE-owned SNF waste package after 10,000 years. For commercial SNF, the lowest waste package retention fraction for fission products (modeled as gadolinium) was calculated as 0.86 (SNL 2007 [DIRS 181165], Tables 8.1-1 and 8.1-2). Table 4.1-11 summarizes miscellaneous criticality input data used in this analysis.

Table 4.1-11. Miscellaneous Direct Inputs

Description	Source
Low criticality potential for DOE-owned SNF canisters fabricated per specification	BSC 2006 [DIRS 181335], Section 7.10
Minimum critical mass of ^{235}U as schoepite	SNL 2007 [DIRS 181395], Table 6.9-1[a], Section 8.1.4[a]

4.2 CRITERIA

This section addresses the criteria relevant to the FEP screening process. These criteria stem from the applicable regulations of 10 CFR Part 63 [DIRS 180319]. These criteria are expanded upon and expressed as specific NRC acceptance criteria in *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274], Sections 2.2.1.1.3, 2.2.1.2.1.3, 2.2.1.2.2.3, 2.2.1.3.1.3, 2.2.1.3.2.3, 2.2.1.3.3.3, 2.2.1.3.4.3, 2.2.1.3.7.3, and 2.2.1.3.9.3).

4.2.1 Yucca Mountain Review Plan (YMRP)

The bases for the NRC review of the license application and its acceptance are described in *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]). The FEP-related acceptance criteria and how this analysis addresses these criteria are presented in Table 4.2-1. The YMRP criteria not relevant to this analysis, as stated in Section 1.1, are YMRP Sections 2.2.1.1.3, 2.2.1.3.1.3, 2.2.1.3.2.3, 2.2.1.3.3.3, 2.2.1.3.4.3, 2.2.1.3.7.3, and 2.2.1.3.9.3 and not addressed in Table 4-2.1. The acceptance criteria for FEP screening justifications rely mainly on the collective screening tests of low probability and low consequence, but also allow for exclusion of a FEP if the process is specifically excluded by the regulations (refer to Section 4.2.2). Note that the criticality FEPs screening justifications rely exclusively on low probability.

Table 4.2-1. Relevant Yucca Mountain Review Plan Acceptance Criteria

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
Scenario Analysis and Event Probability: Scenario Analysis (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3)	**Acceptance Criterion 1:** The Identification of a List of Features, Events, and Processes Is Adequate	(1) The Safety Analysis Report 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 is consistent with the site characterization data. Moreover, the comprehensive features, events, and processes list includes, but is not limited to, potentially disruptive events related to igneous activity (extrusive and intrusive); seismic shaking (high-frequency-low-magnitude, and rare large-magnitude events); tectonic evolution (slip on existing faults and formation of new faults); climatic change (change to pluvial conditions); and criticality.	(1) The list of criticality FEPs and FEP descriptions is provided in Section 1.2. See Section 1.2 for a description and origin of the criticality FEP list. This analysis report does not address climatic change.
Scenario Analysis and Event Probability: Scenario Analysis (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3)	**Acceptance Criterion 2:** Screening of the List of Features, Events, and Processes Is Appropriate	(1) The U.S. Department of Energy 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;	(1) The scenarios important for postclosure criticality have been identified and addressed in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505]) where Section 1.4 describes how this methodology is applied for excluding FEPs.
		(2) The U.S. Department of Energy has provided justification for those features, events, and processes that have been excluded. An acceptable justification for excluding features, events, and processes 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 the regulatory criterion; 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 (low consequence); and	(2) See the method and approach discussion provided in Section 4.2.2 and the individual justification (by low probability, low consequence, or regulation) for excluding FEPs.

Table 4.2-1. Relevant Yucca Mountain Review Plan Acceptance Criteria (Continued)

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
		(3) The U.S. Department of Energy 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 the regulatory criterion; 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.	(3) Sections 6.3 to 6.6 and Table 7.3-1 provide a discussion of the individual FEP screening justifications and supporting technical bases.
Scenario Analysis and Event Probability: Scenario Analysis (NRC 2003 [DIRS 163274], Section	**Acceptance Criterion 3:** Formation of Scenario Classes Using the Reduced Set of Events is Adequate	(1) Scenario classes are mutually exclusive and complete, clearly documented, and technically acceptable.	(1) A comprehensive set of degradation scenarios, identified as the master scenario list, that must be considered as part of the criticality analysis for any waste form has been developed and reviewed as discussed in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2003 [DIRS 165505], Section 3.3).

Table 4.2-1.Relevant Yucca Mountain Review Plan Acceptance Criteria (Continued)

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
2.2.1.2.1.3)	**Acceptance Criterion 4:** Screening of Scenario Classes is Appropriate	(1) Screening of scenario classes is comprehensive, clearly documented, and technically acceptable; (2) The U. S. Department of Energy has adequately considered coupling of processes in estimates of consequences used to screen scenario classes; (3) Scenario classes that are screened from the performance assessment, on the basis that they are specifically ruled out by regulation or are contrary to stated regulatory assumptions are identified, and sufficient justifications are provided; (4) Scenario classes that are screened from the performance assessment, on the basis that their probabilities fall below the regulatory criterion, are identified, and sufficient justifications are provided; and (5) Scenario classes that are screened from the performance assessment, on the basis that their omission would not significantly change the magnitude and time of the resulting radiological exposure to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, are identified, and sufficient justifications are provided.	(1) Screening justifications for the criticality FEPs are summarized in Section 7.1 of this document; (2) The low consequence criterion for screening criticality FEPs was not used in the screening justifications; (3) The application of regulatory rules as a criterion for screening criticality FEPs was not used in the screening justifications; (4) Screening justifications for the criticality FEPs are based on low probability and summarized in Section 7.1 of this document; (5) Since scenario classes associated with criticality FEPs are screened from further consideration in performance assessments, there are no additional radionuclide sources to be considered that could adversely affect radiological exposures to the reasonably maximally exposed individual.

Table 4.2-1. Relevant Yucca Mountain Review Plan Acceptance Criteria (Continued)

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
Scenario Analysis and Event Probability: Identification of Events with Probability Greater than 10^{-8} per Year (NRC 2003 [DIRS 163274], Section 2.2.1.2.2.3)	**Acceptance Criterion 1:** Events Are Adequately Defined	(1) Events or event classes are defined without ambiguity and used consistently in probability models, such that probabilities for each event or event class are estimated separately; and (2) Probabilities of intrusive and extrusive igneous events are calculated separately. Definitions of faulting and earthquakes are derived from the historical record, paleoseismic studies, or geological analyses. Criticality events are calculated separately by location.	(1) See the FEP description provided for each FEP in Table 1-2.1. (2) Probabilities associated with seismic faulting, seismic vibration, and igneous disruptive events are derived from *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828]), *Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion* (SNL 2007 [DIRS 178851]), and *Dike/Drift Interactions* (SNL 2007 [DIRS 177430]), respectively. Analysis associated with the criticality FEPs in Table 1.2-1 address initiating events and location separately in Section 6 of this document.
	Acceptance Criterion 2: Probability Estimates for Future Events Are Supported by Appropriate Technical Bases	(1) Probabilities for future natural events have considered past patterns of the natural events in the Yucca Mountain region, considering the likely future conditions and interactions of the natural and engineered repository system. These probability estimates have specifically included igneous events, faulting and seismic events, and criticality events.	(1) Probabilities associated with seismic and igneous disruptive events (including faulting) are taken into account in the criticality FEPs analyses evaluated separately in Section 6 of this document.

Table 4.2-1.Relevant Yucca Mountain Review Plan Acceptance Criteria (Continued)

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
	Acceptance Criterion 3: Probability Model Support is Adequate	(1) Probability models are justified through comparison with output from detailed process-level models and/or empirical observations (e.g., laboratory testing, field measurements, or natural analogs, including Yucca Mountain site data). Specifically: (a) For infrequent events, the U.S. Department of Energy justifies, to the extent appropriate, proposed probability models with data from reasonably analogous systems. Analog systems should contain significantly more events than the Yucca Mountain system, to provide reasonable evaluations of probability model performance. (b) The U.S. Department of Energy justifies, to the extent appropriate, the ability of probability models to produce results consistent with the timing and characteristics (e.g., location and magnitude) of successive past events in the Yucca Mountain system; and (c) The U.S. Department of Energy probability models for natural events use underlying geologic bases (e.g., tectonic models) that are consistent with other relevant features, events, and processes evaluated, using Section 2.2.1.2.1.	(1) The probability models used in developing parameters for this criticality screening analysis are developed in the appropriate analysis model reports (e.g., *Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion* (SNL 2007 [DIRS 178851]), *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828]), and *Dike/Drift Interactions* SNL 2007 [DIRS 177430])). The validation of the relevant models is documented in the respective reports.

Table 4.2-1.Relevant Yucca Mountain Review Plan Acceptance Criteria (Continued)

YMRP Section	Acceptance Criterion	Description	How Addressed in this Analysis Report
Scenario Analysis and Event Probability: Identification of Events with Probability Greater than 10^{-8} per Year (NRC 2003 [DIRS 163274], Section 2.2.1.2.2.3)	**Acceptance Criterion 4:** Probability Model Parameters Have Been Adequately Established	(1) Parameters used in probability models are technically justified and documented by the U.S. Department of Energy. Specifically: (a) Parameters for probability models are constrained by data from the Yucca Mountain region and engineered repository system to the extent practical; (b) The U.S. Department of Energy appropriately establishes reasonable and consistent correlations between parameters; and (c) Where sufficient data do not exist, the definition of parameter values and conceptual models is based on appropriate use of other sources, such as expert elicitation conducted in accordance with appropriate guidance.	(1) The probability parameters used in this criticality screening analysis are abstracted from appropriate analysis model reports. (a) Parameters for the probability evaluations used in this analysis are derived from the output of the validated models for the Yucca Mountain repository. (b) Parameter correlations, if any, are justified in the referenced reports. (c) No assumptions are used in this analysis.
	Acceptance Criterion 5: Uncertainty in Event Probability Is Adequately Evaluated	(1) Probability values appropriately reflect uncertainties. Specifically: (a) The U.S. Department of Energy provides a technical basis for probability values used, and the values account for the uncertainty in the probability estimates; and (b) The uncertainty for reported probability values adequately reflects the influence of parameter uncertainty on the range of model results (i.e., precision) and the model uncertainty, as it affects the timing and magnitude of past events (i.e., accuracy).	(1) Uncertainty in probability values is acknowledged in this report but not quantified. The uncertainty in model outputs used to develop probabilities is discussed in the respective reports. Specifically: (a) A technical basis for probability values used in this analysis is provided in Sections 6.3 through 6.6. (b) The mean values of parameters from distributions are used but uncertainties are not reported separately for probability values. The probability values are based on results that incorporate the parameter uncertainty from the model results and model uncertainty.

4.2.2 FEPs Screening Criteria

The criteria for determining low probability, low consequence, or by regulation exclusion are described in the following sections.

4.2.2.1 Low Probability

The low-probability criterion is stated in 10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53318, Section 114(a)(4)) (proposed rule) requires any performance assessment used to demonstrate compliance with Part 63 Section 113 for 10,000 years after disposal to:

> Consider only features, events, and processes consistent with the limits on performance assessment specified at § 63.342.

Where 10 CFR Part 63 Section 342(a) (proposed rule) requires:

> DOE's performance assessments conducted to show compliance with § 63.311(a)(1), § 63.321(b)(1), and 10 CFR 62.331 shall not include consideration of very unlikely features, events, and processes, i.e., those that are estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal (70 FR 53313 [DIRS 178394], pp. 53319 and 53320).

For time-dependent FEPs, such as criticality, 10 CFR Part 63 (70 FR 53313 [DIRS 178394], p. 53319, Section 342(a)) (proposed rule) is also expressed as a total probability criterion equivalent to a probability of occurrence of 10^{-4} over 10,000 years.

4.2.2.2 Low Consequence

The low consequence criteria as stated in 10 CFR Part 63 (70 FR 53313 [DIRS 178394], pp. 53318 to 53319, Section 114(a)(5) and (6)) (proposed rule), any performance assessment used to demonstrate compliance with 10 CFR 63.113 for 10,000 years after disposal must:

(a)(5) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes in the performance assessment. Specific features, events, and processes must be evaluated in detail 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.

(a)(6) Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail 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

Additional support for the low consequence screening criterion is provided in 10 CFR Part 63 [DIRS 180319], Subpart E, Section 113 (b) and (c), which states:

(b) The engineered barrier system must be designed so that, working in combination with natural barriers, radiological exposures to the reasonably maximally exposed individual are within the limits specified at § 63.311 of subpart L of this part. Compliance with this paragraph must be demonstrated through a performance assessment that meets the requirements specified at § 63.114 of this subpart, and §§ 63.303, 63.305, 63.312, and 63.342 of subpart L of this part.

(c) The engineered barrier system must be designed so that, working in combination with natural barriers, releases of radionuclides into the accessible environment are within the limits specified at § 63.331 of subpart L of this part. Compliance with this paragraph must be demonstrated through a performance assessment that meets the requirements specified at § 63.114 of this subpart and §§ 63.303, 63.332 and 63.342 of subpart L of this part.

Some FEPs have a beneficial effect on the TSPA, as opposed to an adverse effect. As identified in 10 CFR Part 63 [DIRS 180319], Section 102(j), the concept of a performance assessment includes:

> The features, events, and processes considered in the performance assessment should represent a wide range of both beneficial and potentially adverse effects on performance (e.g., beneficial effects of radionuclide sorption; potentially adverse effects of fracture flow or a criticality event). Those features, events, and processes expected to materially affect compliance with 10 CFR Part 63 {[DIRS 180319], Section 113(b)} or be potentially adverse to performance are included, while events (event classes or scenario classes) that are very unlikely (less than one chance in 10,000 over 10,000 years) can be excluded from the analysis…

Yucca Mountain Review Plan, Final Report (NRC 2003 [DIRS 163274], Section 2.2.1), states:

> In many regulatory applications, a conservative approach can be used to decrease the need to collect additional information or to justify a simplified modeling approach. Conservative estimates for the dose to the reasonably maximally exposed individual may be used to demonstrate that the proposed repository meets U.S. Nuclear Regulatory Commission regulations and provides adequate protection of public health and safety. …The total system performance assessment is a complex analysis with many parameters, and the U.S. Department of Energy may use conservative assumptions to simplify its approaches and data collection needs. However, a technical basis … must be provided.

It is preferable to include beneficial FEPs in TSPA-LA. However, in some cases a beneficial FEP may not be implemented in TSPA-LA (e.g., where there is an insufficient technical basis for inclusion). In these cases, it is acceptable, based on the above statements, to demonstrate that a beneficial FEP can only improve the performance (of an otherwise compliant system) and therefore that its omission cannot "materially affect compliance." In these cases, FEPs that are demonstrated to have only beneficial effects on the radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, can be excluded on the basis of low consequence because they have no adverse effects on performance.

4.2.2.3 Regulation

Yucca Mountain Review Plan, Final Report (NRC 2003 [DIRS 163274], Section 2.2.1.2.1.3, Acceptance Criterion 2) allows for exclusion of a FEP if the process is specifically excluded by the regulations. To wit:

> The DOE has provided justification for those FEPs that have been excluded. An acceptable justification for excluding FEPs is that either *the FEP is specifically excluded by regulation*; probability of the FEP (generally an event) falls below the regulatory criterion; 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.

Furthermore, the *Public Health and Environmental Protection Standards for Yucca Mountain, Nevada* (40 CFR Part 197 [DIRS 184076], Subpart B, Section 36) (proposed standard) states:

> …The DOE's performance assessments conducted to show compliance with §§ 197.20(a)(1), 197.25(b)(1), and 197.30 shall not include consideration of very unlikely features, events, and processes…DOE's performance assessments need not evaluate the impacts resulting from any features, events, and processes or sequences of events and processes with a higher chance of occurrence if the results of the performance assessments would not be changed significantly in the initial 10,000-year period after disposal.

Thus, evaluation of the criticality FEPs need not extend past the 10,000-year period provided criticality can be screened from consideration in TSPA.

4.3 CODES AND STANDARDS

The following codes are cited in this analysis:

- 40 CFR 197 2005 [DIRS 184076]. Protection of Environment: Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada. ACC: MOL.20051121.0084.

- 70 FR 53313 [DIRS 178394]. Implementation of a Dose Standard After 10,000 Years. Internet Accessible.

- 10 CFR 63 2007 [DIRS 180319]. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. Internet Accessible.

- ASTM B 932-04 2004 [DIRS 168403]. *Standard Specification for Low-Carbon Nickel-Chromium-Molybdenum-Gadolinium Alloy Plate, Sheet, and Strip.* West Conshohocken, Pennsylvania: American Society for Testing and Materials.

- ASTM A 887-89 (Reapproved 2004). 2004 [DIRS 178058]. *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application.* West Conshohocken, Pennsylvania: American Society for Testing and Materials.

INTENTIONALLY LEFT BLANK

5. ASSUMPTIONS

No general assumptions are used in the development of inputs for the probability of criticality evaluations.

INTENTIONALLY LEFT BLANK

6. SCIENTIFIC ANALYSIS DISCUSSION

6.1 PROBABILITY OF CRITICALITY CALCULATIONAL APPROACH

The following sections discuss the processes used in evaluating the probability of occurrence of configurations in the repository with potential for criticality. Section 6.2 discusses the approach to organizing the processes and event scenarios. Section 6.3 provides the details for the criticality screening justifications for the early failure event FEP scenarios (all criticality FEP scenarios are listed in Table 1.2-1). Section 6.4 provides the details for the disruptive seismic event FEP scenarios, Section 6.5 for the single block rockfall disruptive event FEP scenarios, and Section 6.6 for the igneous disruptive event FEP scenarios.

6.2 SCENARIOS IMPORTANT FOR CRITICALITY

During design, criticality analyses are performed to demonstrate that the initial emplaced configuration of the waste form remains subcritical, even under flooded conditions. Several potential configurations that could occur in the repository over the 10,000-year regulatory period are selected, based on sensitivity studies, in the development of the loading curves that result in the highest k_{eff} in order to set an upper bounding limit that encompasses all other configurations. The loading curve is established such that the k_{eff} of a waste package fully loaded with assemblies selected from the curve will be less than a certain critical limit under all postulated postclosure conditions (SNL 2008 [DIRS 182788], Section 1). The design basis configuration developed in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008 [DIRS 182788], Section 6.2.4.1) is considered to bound the various limiting configurations that would result for each of the criticality FEP scenarios (early failure of engineered barrier, seismic, rockfall, and igneous) for commercial SNF. The nominal FEP scenarios address non-disruptive events that can affect the probability of criticality for the repository. The seismic FEP scenarios include both vibratory and faulting events. The vibratory seismic scenarios include most rockfall events since a seismic event is the initiator for such events that can affect the probability of criticality for the repository. The rockfall FEP scenario is limited to nominal rockfall events not caused by seismic activity that could damage a waste package.

For a criticality event to occur, the appropriate combination of materials (e.g., neutron moderators, neutron absorbers, fissile materials, or isotopes) and geometric configurations favorable to criticality must exist. Therefore, for a configuration to have potential for criticality, all of the following conditions must occur: (1) sufficient mechanical or corrosive damage to the waste package OCB to cause a breach, (2) presence of a moderator (i.e., water), (3) separation of fissionable material from the neutron absorber material or an absorber material selection error during the canister fabrication process, and (4) the accumulation (external) or presence of a critical mass of fissionable material. The probability of developing a configuration with criticality potential is insignificant unless all four conditions are realized, and then is only representative of a conservative estimate since the probability values associated with the many other events required to generate a critical configuration have been conservatively set to one (1).

The National Spent Nuclear Fuel Program has categorized the DOE-owned SNF inventory into groups based on fuel matrix, cladding, cladding condition, and enrichment to support both TSPA and criticality analyses. The representative fuel in each condensed group (Table 4.1-2) was selected based generally on the quantity of the SNF within that specific group (Radulescu et al. 2004 [DIRS 165482], Section 3.1.1).

As stated in Section 1, an evaluation of the criticality FEP scenarios for configurations with potential for criticality has identified sets of events that are common to each of the sequence of events within the scenario classes that must occur for a criticality event to occur. Two independent sets of events for the in-package scenarios were identified (i.e., material selection errors for the neutron absorber material during canister fabrication processes and, for pressurized water reactor canisters, waste form misloads due to loading curve violations) for which the estimated probability of occurrence is central to the FEPs screening justifications. These sequences of events coupled with the initiating event are evaluated in the following sections to provide a conservative estimate for the probability of achieving a configuration with potential for criticality. This is a conservative estimate because the probability of criticality for a misload event is a combination of the probability that a misload event occurs and the bounding design basis configuration that maximizes the criticality potential. Therefore, the actual limiting configuration for the scenario class would have less influence on the maximum k_{eff} resulting from a misload (i.e. a less reactive configuration) and, secondly, not all assemblies within the package being of the most reactive assembly type and thus the worst case (highest criticality potential) assembly loading arrangement. Since the two misload events (waste form and absorber plate) are independent and either event can create a critical condition, the end probability values are additive. The input probabilities used for the probability estimates are either a mean or a point estimate value. In addition, as stated in Section 1, the overall probability of a sequence of events is a monotonically decreasing value as additional events are included to generate a full sequence (BSC 2004 [DIRS 172494], Appendix I). These additional events would be ascribed to those necessary to form the actual configuration being considered, or a potentially more reactive one. Analyzing these extended sequences of events would result in probabilities of attaining the configurations that are less than those estimated from this analysis since the probabilities of each additional event is at most one and, for most events, much less than one.

The presence of neutron absorber materials in waste package canisters is important for criticality control during the 10,000-year period following repository closure for the majority of the canisters proposed for disposal. Note that evaluation of criticality FEPs need not extend past the initial 10,000-year period following repository closure if criticality can be screened from consideration during that period (Section 4.2.2.3). Neutron absorber material misloads can occur in TAD canisters as the result of various operations (or the lack thereof) during the canister fabrication and loading processes. These processes or operations that could adversely affect the criticality potential of waste forms include the use of wrong materials during fabrication and/or failure to properly load the neutron absorber materials into the canister.

Detailed criticality analyses for screening justifications have been completed for a subset of the waste package and waste form combinations for license application. This subset is composed of the 21-PWR, 44-BWR, and a representative fuel type for each of the nine DOE-owned SNF groups (Table 4.1-2). To accommodate pressurized water reactor SNF with characteristics outside waste form loading curves, specifications for the TAD canister design require the capability of loading fuel assemblies containing disposal control rod assemblies (SNL 2007

[DIRS 179394], Section 4.1.1.5). The fuel type selected for each of the nine DOE-owned SNF groups represents the characteristics of all fuel types in that group (Radulescu et al. 2004 [DIRS 165482], Executive Summary). Representative configurations for the degradation and reconfiguration of the internal structures and waste forms in a waste package have been addressed in numerous analyses for the various SNF fuel types. These results from Section 4.1.15 are summarized in *CSNF Loading Curve Sensitivity Analysis* (SNL 2008 [DIRS 182788]), *DOE SNF Phase I and II Summary Report* (BSC 2004 [DIRS 165482]), *Intact and Degraded Mode Criticality Calculations for the Codisposal of TMI-2 Spent Nuclear Fuel in a Waste Package* (BSC 2004 [DIRS 168935]), and *Intact and Degraded Mode Criticality Calculations for the Codisposal of ATR Spent Nuclear Fuel in a Waste Package* (BSC 2004 [DIRS 171926]). The results indicated that the maximum k_{eff} of the various configurations were less than the critical limit. For selected DOE-owned SNF waste forms, the neutron absorber material is an integral part of the waste form as the absorber material is integrated into the DOE standardized SNF canister in the form of shot at the time of waste form loading per Section 5.2.3 of *Criticality Potential of Intact DOE SNF Canisters in a Misloaded Dry Waste Package* cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for DOE SNF/HLW and Naval SNF Waste Package Physical Attributes Basis for Performance Assessment* (SNL 2007 [DIRS 179567], Table 4-1, Item 03-02).

The process controls for loading DOE-owned SNF canisters are expected to be similar to NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* and, since DOE-owned SNF canisters must be shipped to the repository, the quality assurance requirements of 10 CFR Part 71 [DIRS 173375], Subpart H, must be met. Thus, sufficiently rigorous requirements are expected to be in place to reduce the likelihood of accepting canisters without the shot type of absorber material to insignificant values. A possible second quality check for a lack of shot type absorber in a canister is a weight measurement of the loaded canister since such errors can be readily detected. The potential for criticality in DOE-owned SNF canisters using shot as an absorber is considered insignificant and not analyzed further. The DOE-owned SNF waste forms that require plate type neutron absorber materials (Ni-Gd) from Section 4.1.15 are MOX (DOE1), $UZrH_x$ (DOE2), and aluminum-based SNF (DOE7). Errors in canister fabrication that result in an improper performance of the neutron absorber material are captured in the configuration generator model report (BSC 2004 [DIRS 172494], Figure I-5) as Top Events "NA-MISLOAD" (neutron absorber-misload). Such a misload of the absorber material in a DOE-owned SNF canister is possible during canister fabrication from installation of material not meeting specifications and the probability of such an event is evaluated in a similar manner as absorber misloads for commercial SNF canisters.

Scenarios important for criticality in the 10,000-year period following repository closure primarily result from potential deviations from the design configuration in the fabrication and loading processes prior to shipment to the repository. For commercial SNF, analyses demonstrate that intact, fully flooded with water (i.e., a neutron moderator) TAD canister waste package configurations as designed will not achieve criticality (Section 4.1.5). In addition, commercial and DOE-owned SNF canisters must meet requirements for handling, transportation, and storage of fissile material as specified, for example, in ANSI/ANS-8.17-2004 [DIRS 176225] or Regulatory Guide 3.71 [DIRS 176331]. These requirements specify criteria for establishing subcritical configurations in the transportation and storage containers. Demonstration that these criteria have been met, a necessary requirement for shipment, will provide additional assurance that the probability of achieving configurations with potential for criticality in the postclosure period are sufficiently low to be screened from consideration in performance analyses.

Several of the scenario evaluations include additional unquantified conservatisms as no credit is taken for the stainless steel liner or TAD canister in the commercial SNF waste packages or for the DOE-owned SNF canister in the codisposal waste packages as a barrier to water ingress. All of the internal waste package components are considered to fail when the waste package OCB is breached.

6.2.1 In-Package Scenarios

As stated in Section 6.2, the waste package/waste form configuration must degrade or deviate in some manner from the design configuration to achieve a potentially critical configuration. This is because the as-designed intact commercial SNF waste package in a fully flooded environment is precluded from achieving criticality (SNL 2008 [DIRS 182788], Figure 6-6). Likewise, criticality evaluations for DOE-owned SNF include flooded conditions (Radulescu et al. 2004 [DIRS 165482], Section 10). Even if a waste package is breached, the very low corrosion rates of the waste package materials (Section 4.1.14) effectively prevent potentially critical configurations from developing over the regulatory period by internal reconfigurations that separate fissile material from absorber material. Deviations from the design configuration could result from undetected operational failures (e.g., fabrication processes, waste form loading errors, and drying procedures). The only identified events that can breach a waste package in the early failure scenario during the regulatory period are: (1) stress corrosion cracking initiated from manufacturing defects, (2) misplaced drip shields allowing advective seepage onto waste packages leading to breaching from localized corrosion, or (3) a deflagration event resulting from radiolytic gas generation and ignited by metal-to-metal motions such as may occur during a non-disruptive seismic event. The TAD canisters (SNL 2007 [DIRS 179394], Table 4-1, Item 04-04) and possibly others (i.e., DOE-owned SNF canisters) are expected to be loaded in spent fuel pools. Intact TAD and DOE canisters and waste packages are expected to contain little moisture per requirements for drying (SNL 2007 [DIRS 179394], Table 4-1, Item 04-04) but retention of water in a canister waste package could possibly occur if the drying and inerting process is incomplete. The process controls for the drying and inerting process for commercial SNF canisters and waste packages are expected to be similar to NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (SNL 2007 [DIRS 179394], Table 4-1, Item 04-04) and, thus, sufficiently rigorous to reduce the likelihood of leaving residual water in the TAD canisters to levels, which, if quantified, would not significantly increase the overall probability of criticality in the repository. The consequences of a deflagration event are discussed in Appendix I as a defense-in-depth contribution without quantitative evaluation of event sequence probabilities.

Fabrication defects in the waste package OCB that can lead to stress corrosion cracking have been analyzed in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007 [DIRS 178765], Section 6). Such events include, for example, improper material selection, improper heat treatment, and waste package OCB lid closure weld flaws. The probabilities associated with the set of fabrication defects in the waste package OCB has been evaluated individually from the respective event tree/fault tree diagrams and collectively with the exception of the weld flaws in the latter case. Thus, probability values from the collective evaluation are used, for example, in the early failure scenario where a waste package OCB breach could result from any one of several SCC initiators. Probability values from the individual evaluations are used where a waste package OCB breach could result from a specific SCC initiator such as failure of the low plasticity burnishing process for stress mitigation.

If seepage is predicted to occur, water in the invert (most likely through failed drip shields) could provide a transport mechanism for degraded fissile material to accumulate in the external environment in the event of waste package breach and subsequent release of the waste form. Water in the invert may also mix with the waste package effluent producing a change in the chemistry that causes the deposition and accumulation of fissile material in the near-field location. Seepage is predicted to vary over the repository depending on several factors (DTN: LB0407AMRU0120.001 [DIRS 173280], file: *Summary_seepage_abstraction.doc*). One of the variables derived from the seepage probability distribution is the fraction of waste packages hit by seepage. Thus, the probability per package of seepage water being available is less than one. However, the relative humidity in the drifts is expected to approach 90% to 100% (SNL 2008 [DIRS 184433], Figure 6.3-68) and, thus, the probability of water in liquid or vapor form being present is conservatively set to 1.0 for the screening justifications in this document. The diffusive flow of humid air into waste packages, and thus accumulation within the waste package, will be limited by outward flow since the interior of the waste packages will be warmer than the external environment for a considerable time period (SNL 2008 [DIRS 179962], Figure 6.4.2-4b).

The accumulation and retention of water in a breached waste package is referred to as a bathtub configuration. If the drip shield fails, it is conceivable that water could enter the waste package, but not accumulate due to a breach in the waste package bottom. This condition is referred to as a flow-through configuration. High relative humidity in the repository drifts also allows water vapor access to the internal structures and waste forms in breached waste packages regardless of the breach location. Potentially critical configurations (e.g., formation of schoepite-moderated systems) could result from such conditions through the degradation of the waste package internals and the separation or removal of neutron absorber and/or fissile materials.

Evaluation of the neutron absorber material misload failure mechanism is an important consideration for the determination of a configuration’s criticality potential. The probability that proper neutron absorber material is not used in the waste package (or waste form if integrally connected) or becomes separated from the fissile material must then be evaluated for configurations where absorber material is necessary for criticality control. Misloading of the waste forms is also an important consideration for the determination of a configuration’s criticality potential for commercial SNF that require restricted loading configurations (i.e., specified loading curves). The probability that such waste forms are not loaded as required must then be evaluated.

6.2.2 External (Near-Field and Far-Field) Scenarios

The probability of external criticality in the near-field is less than the probability of criticality for in-package locations. This is because, in addition to the events evaluated to calculate the probability for external accumulation of fissile material that require a means of transport, the following events or processes must also be considered:

- Separation of the fissile materials from the degraded waste form
- Sufficient seepage water to transport fissile materials to an accumulation site in the external environment
- Reducing environment
- Presence of sufficient neutron moderator.

If the likelihood of having an adequate accumulation of fissile material in the near-field environment to support criticality is determined to be sufficiently low such that, if quantified, would not significantly increase the overall probability of criticality in the repository, then the probability of achieving a far-field critical configuration would be even lower.

The criticality potential in either the near- or far-field environment is insignificant unless the possibility exists for at least the accumulation of sufficient fissionable material to form an optimal critical configuration. The minimum mass that must be accumulated in the invert to achieve criticality, based on a critical limit of 0.96 for all the cases evaluated has been calculated for several different waste forms in *Geochemistry Validation Report: External Accumulation Model* (SNL 2007 [DIRS 181395], Table 6.9-1[a]). The results from this calculation are summarized in Section 4.1.15, which shows that the largest fissile mass for those waste forms that can accumulate in the invert under a breached waste package over a 10,000-year period is much less than the minimum critical mass necessary for criticality in the external locations. Although there is uncertainty in the calculations, this mass difference, coupled with the discussion on the criticality potential of waste forms in Section 4.1.15, is sufficient to consider that the unquantified likelihood of forming either a near- or far-field configuration with criticality potential is insignificant. Thus, the probability of accumulating a critical configuration external to the waste packages is concluded to be insignificant.

6.3 FEPS ASSOCIATED WITH NOMINAL (EARLY FAILURE) EVENT SEQUENCE INITIATORS

This probability of criticality calculation for the postclosure criticality scenarios associated with nominal (early failure) events evaluates the probability that water is able to enter a waste package, either in the vapor or liquid phase, to degrade the waste package internals and waste form. Such an environment is considered as conducive for creating a configuration with criticality potential during the initial 10,000-year period following repository closure in the absence of disruptive events (Table 1.2-1). The probability of achieving potentially critical configurations is considered for both internal and external waste package locations.

Without water infiltration, configurations with criticality potential are very improbable in the repository since it is improbable that a critical mass for unmoderated or silica moderated systems can be accumulated (YMP 2003 [DIRS 165505], Section 3.7.2). All postclosure criticality FEP scenarios, internal and external, require water infiltration in liquid or vapor form to be present to provide adequate moderation to support criticality and to provide a mechanism to degrade the waste package internals and/or the waste form, as intact configurations are subcritical by design (SNL 2007 [DIRS 178236], Section 3.1).

A lack of neutron absorber material from either loss or fabrication defects coupled with sufficient water for neutron moderation is the most reasonable scenario that could result in a potentially critical configuration in any of the in situ criticality FEP scenarios. Seepage flow-through and humid air conditions internal to the waste package may also degrade waste package internal components and waste forms. External criticality FEPs (near-field and far-field) also require the separation of neutron absorber materials from the waste form and, additionally, the transport of fissile material from the waste package and its re-accumulation in the drift invert or beyond.

Water, silica, and carbon are the only potential moderating materials for internal and external configurations. Water, the most effective neutron-moderating material, can enter a breached waste package as seepage flow or as humid air. Silica is present in appreciable quantities in the high-level radioactive waste glass canisters and in the repository rock. Silica can also be introduced into the waste package through precipitation from the seepage flow, given a failed drip shield. Carbon is present but only in less than 20% of the DOE-owned SNF waste package variants (with the exception of fuel type DOE6, Table 4.1-2) and these have no potential for criticality as additional absorber material in the canister is not necessary for criticality control (Radulescu et al. 2004 [DIRS 165482], Section 10.8.6). Furthermore, there is no known mechanism for lateral transport of any carbon in the invert to alternate accumulation sites. Thus, carbon has a limited impact on the over-all potential for criticality in the repository.

Although silica can act as a neutron reflector, inside the waste package its reflector effects that act to increase the k_{eff} are secondary to its water displacement effects that act to decrease the k_{eff} of the system (YMP 2003 [DIRS 165505], Section 3.7.2). Silica moderation from degraded glass in DOE-owned SNF codisposal waste packages is taken into account in the DOE-standardized SNF canisters with the loading design, the basket structure design inside the canisters, and additional neutron absorber materials. Silica moderation from the degradation of high-level radioactive waste glass, therefore, has no impact on the potential for criticality in DOE-owned SNF waste packages. Silica from seepage infiltration will displace water and effectively reduce the criticality potential of the system (since silica is a much less effective moderator than water), thus reducing the potential for criticality in commercial SNF waste packages.

Certain DOE-owned SNF waste forms have sufficient quantities of fissile material to support unmoderated (fast) criticality if the fissile material is concentrated beyond its design concentration in the waste form and the neutron absorber materials are removed. While concentration of the fissile material beyond its nominal design concentration could result from degradation of the waste form by either water infiltration or a disruptive event, removal of the neutron absorber materials from a DOE-owned SNF waste package would require a breach of the waste package and a removal mechanism. Degradation in the presence of water would result in a moderated system since schoepite would likely be the end state of the fissile material. Likewise, there is no known mechanism that could reconfigure non-degraded fissile material into a compact configuration with unmoderated criticality potential. The most likely neutron absorber material removal mechanism is through water infiltration resulting in degradation of the waste package internal components, dissolving of the neutron absorber material in the water, and flushing of the material from the waste package. This mechanism is unlikely to result in a critical configuration since the corrosion rate of the Ni-Gd neutron absorber material is very low (Table 4.1-8) with < 6 mm (likely much less) removed over the 10,000-year period. In addition, the gadolinium in the DOE-owned SNF canisters forms phosphate or carbonate corrosion products (SNL 2007 [DIRS 181165], Section 6.3.16), which have very low corrosion rates.

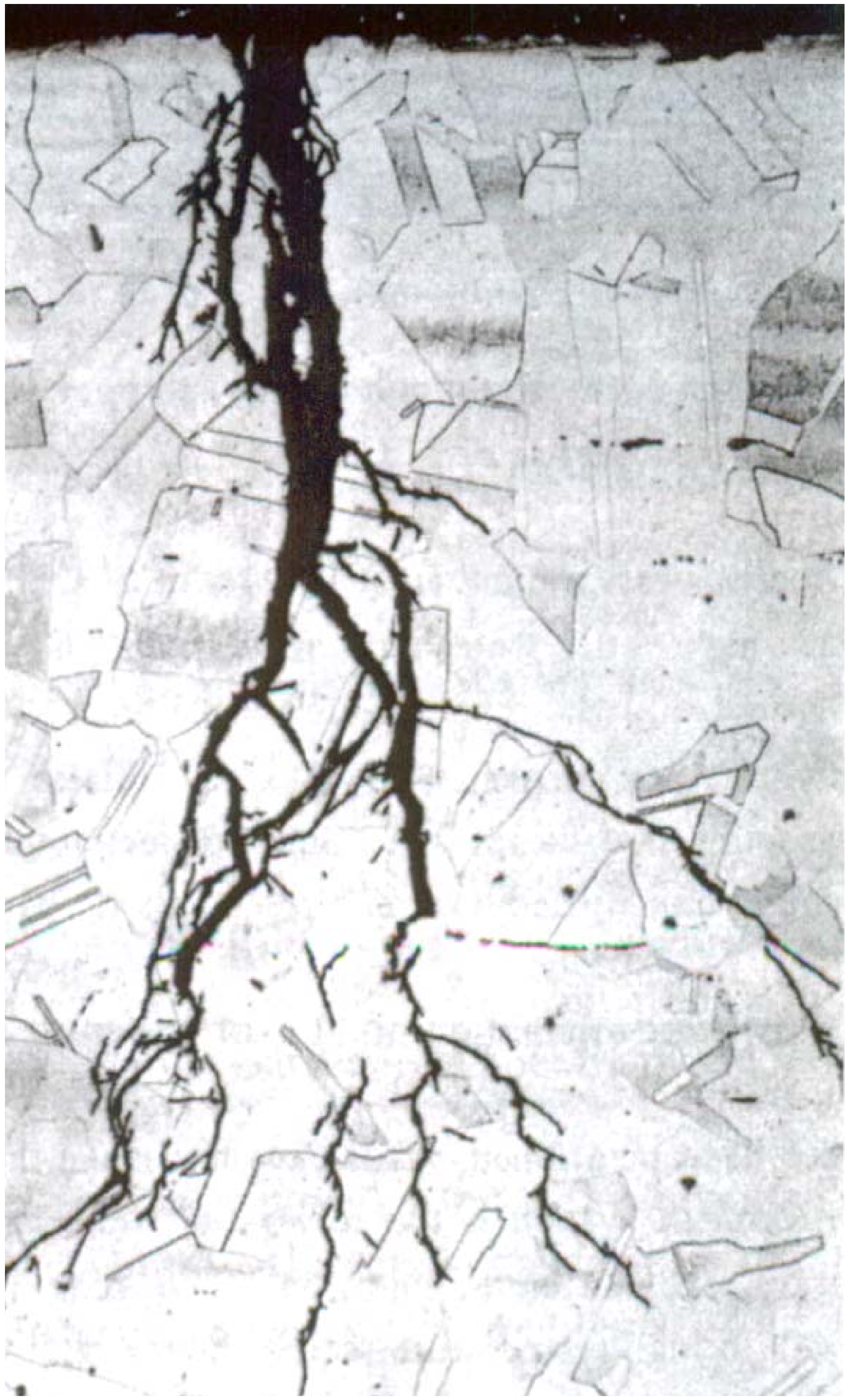
The neutron absorber material within canisters is the primary mechanism for criticality control throughout the postclosure era. This material must be able to maintain its function under long-term exposure to environments with varying levels of corrosive potential, mechanical disruption from seismic events, and immersion in high temperature magmatic environments. The very low corrosion rates of the Ni-Gd alloy absorber material (Table 4.1-8) proposed for use in the DOE-owned SNF canisters effectively limit the absorber loss, given a waste package breach, to at most a few millimeters as stated previously. Thus, the estimated probability of a criticality developing from this sequence of events is sufficiently low such that, if quantified, it would not significantly increase the overall probability of criticality in the repository (Table 4.1-10).

The absorber material designated for the TAD canisters is borated stainless steel (Table 4.1-8) produced by powder metallurgy that results in a near-optimal dispersion of boron throughout the material (ASTM A 887-89 Grade A [DIRS 178058], pp. 1 to 4). This material has acceptable long-term neutron control characteristics based upon the near uniform particle dispersion in the absorber material together with acceptable corrosion behavior as extrapolated from short-term exposure tests. The corrosion rates from Table 4.1-8 indicate that absorber loss from corrosion, given a waste package breach, is expected to be on the order of millimeters or less over the first 10,000 years after emplacement.

The early failure event criticality FEP scenarios are identified as FEPs 2.1.14.15.0A, 2.1.14.16.0A, 2.1.14.17.0A, and 2.2.14.09.0A (Table 1.2-1). The scenarios associated with FEPs are: (1) in-package criticality (intact configuration), (2) in-package criticality (degraded configuration), (3) near-field criticality, and (4) far-field criticality. The early failure event FEP scenarios incorporate three locations. As noted in Section 1.2, the two in-package locations (intact and degraded) are essentially the same since the degraded in-package configurations differ from the intact configurations primarily in the waste form composition, once the waste package is breached. The same initial set of events must occur in the scenarios for the in-package location prior to splitting into separate sequences that lead to configurations with potential for criticality. The sequences evaluated in this analysis have been truncated prior to their split for evaluating events peculiar to the intact and degraded locations, respectively. Therefore, only the set of events for the in-package location associated with the degraded scenarios have been selected for evaluation. This applies to all of the in-package FEPs, disruptive as well as early failure of engineered barrier events, in this analysis.

The criticality potential of the in-package intact configuration scenario is negligible since intact configurations, fabricated and loaded according to specifications, will remain subcritical given that any degradation of intact configurations is insufficient by design to compromise the functioning of criticality control structures (Section 1.2). Although configurations not conforming to design specifications are applicable to both intact and degraded scenarios, configurations with potential for criticality require sufficient water for moderation. Since the internals are degraded for configurations in the in-package degraded scenario, the cladding is considered breached within the failed waste package and the interior of the fuel rods are assumed to be exposed to the repository environment allowing the fissile material to convert to schoepite. The criticality potential of the in-package degraded configuration scenario is negligible provided that waste packages are fabricated and loaded according to design specifications as sensitivity studies have shown that the pressurized water reactor SNF waste form in various degraded configurations, such as saturated porous schoepite, does not result in a more reactive configuration than the design basis configuration (SNL 2007 [DIRS 181373], Table A-12; SNL 2008 [DIRS 182788], Section 6.2.5).

The only such identified events that can breach a waste package in the early failure event over the regulatory period are stress corrosion cracking initiated from either weld flaws in the waste package OCB lid or undetected fabrication defects in the waste package OCB, and improperly emplaced drip shields that allow advective flow onto the waste package OCB, which may permit localized corrosion to develop. Stress corrosion cracking of the waste package OCB is addressed as an *included* FEP in FEP 2.1.03.02.0A (SNL 2007 [DIRS 179476], Section 2.1.03.02.0A) but requires an initiator mechanism such as mechanical damage except for weld flaws in the OCB closure lid where residual tensile stresses can exist. Even if a waste package were to fail early because of a defect, only a limited amount of water could collect in the waste package. This is because most through-wall penetrations, especially cracks from stress corrosion cracking, are usually tight and of limited length based on observations of SCC morphology in Alloy 22, which is expected to be transgranular rather than intergranular, as commonly observed in high-tensile environments such as light water reactors (Herrera 2004 [DIRS 168133], Section 2.0). A typical example of transgranular type SCCs is illustrated in Figure 6.3-1 for stainless steel. (While Figure 6.3-1 does not have an embedded length scale, typical SCC opening widths range from 0.01 to 0.05 mm (SNL 2007 [DIRS 181953], Section 6.3.3).) Note that no credit is taken for the reduction in the rate of water ingress into a failed waste package due to the presence of the stainless steel inner liner or, for commercial SNF, the TAD canister and for codisposal, the DOE-owned SNF canister.



Source: Herrera 2004 [DIRS 168133], Figure 2-1.

Figure 6.3-1. Typical Example of Transgranular Stress Corrosion Cracking Cracks in Stainless Steel

A bounding calculation for the rate of water transport through cracks in both the drip shield and waste package under seepage drips indicates that the maximum volumetric flow rate through a crack or cracks into a given waste package is very low (DTN: SN0705WFLOWSCC.001 [DIRS 184848], file: *Analysis for Water Flow through Stress Corrosion Cracking (SCC) Cracks in Waste Package and Drip Shield.xls*, spreadsheets "Sheet flow DS flow rate" and "Sheet flow

WP flow rate") and insufficient to support localized corrosion. However, an improperly emplaced drip shield could result in an advective flow that could support localized corrosion. In addition, the interior of the waste packages will be warmer than the external environment for a considerable time period (SNL 2008 [DIRS 179962], Figure 6.4.2-4b). The accumulation of water within the waste package will be limited by evaporation through the breeches because of the warmer waste package as discussed in *Waste Package Flooding Probability Due to Seismic Fault Displacement* (SNL 2008 [DIRS 184078], Section 6.2.1.4). In addition, the intact configuration is designed to remain subcritical when fully flooded (SNL 2008 [DIRS 182788], Figure 6-6) and the design basis configuration accounts for corrosion loss of the neutron absorber over the first 10,000-year period following repository closure. Likewise, significant geometrical reconfigurations would be very improbable from waste package breaches that are limited to SCC or localized corrosion since the internal structure remains in place.

Seismic analyses (SNL 2007 [DIRS 176828], Section 6.7.3.1), however, have indicated that the drip shield will be partly surrounded by rockfall at PGV levels that are below the levels with the potential for causing separation, and this rockfall occurs within the first few seconds of the ground motion. The larger rock blocks or the lithophysal rubble provide normal and shear confinement to the sidewalls and possibly the crown of the drip shield. The horizontal acceleration imparted to the drip shield by the ground motion will be resisted by the weight of the rockfall and by the frictional forces between the rock and the drip shield plates and between the footings and the invert. The exterior bulkheads on the sidewalls of the drip shield provide an additional physical restraint or "locking" mechanism between the drip shield and rubble that will constrain axial movement. Thus, the presence of rockfall around the drip shields will restrict the relative displacements that are required to separate adjacent drip shields, so that separation is not expected to occur, even for extreme ground motions. Thus, it is very improbable that a waste package will be exposed to the maximum seepage rate associated with the drip shield loss of function except for improperly emplaced drip shields.

Therefore, stress corrosion cracking in the OCB closure lid welds of waste packages is the most credible (but not the sole) initiator for events in the early failure of engineered barrier criticality FEP scenario that, coupled with neutron absorber material misload events (Section 6.2) and, for 21-PWR TAD canisters, with waste form misload events, could lead to configurations with potential for criticality.

A waste package breach is not expected to increase the criticality potential for the near-field location, or for the far-field location, (FEPs 2.1.14.17.0A and 2.2.14.09.0A, respectively, for the early failure event, Table 1.2-1). Section 6.2.2 discusses the minimum fissile mass necessary for criticality external to the waste packages (Tables 4.1-9 and 4.1-10) where it is concluded that insufficient fissile material can collect over the first 10,000-year postclosure period to achieve a critical mass.

6.3.1 Stress Corrosion Cracking in the OCB Closure Lid Welds

Sources of corrosion of the waste package OCB have been considered in the screening of processes affecting waste package degradation in *Features, Events, and Processes for the Total System Performance Assessment* (SNL 2007 [DIRS 179476], FEP 2.1.03.02.0A). This FEP identifies the propagation of incipient cracks that can occur on the waste package outer barrier closure welds (since these cannot be annealed to relieve tensile stress but stress mitigation processes will be employed (i.e., low plasticity burnishing)) and/or fabrication flaws in the waste packages as possible initiating mechanisms for the development of stress corrosion cracking of the waste package outer barrier. These mechanisms are discussed in *Stress Corrosion Cracking of Waste Package Outer Barrier and Drip Shield Materials* (SNL 2007 [DIRS 181953], Section 8.4.2.1) and *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007 [DIRS 178765], Section 6.3). Stress corrosion cracks can be initiated on a smooth weld surface (with incipient cracks) or at existing weld flaws if the tensile stresses exceed the threshold stress for SCC nucleation that is taken to be 90% to 105% of the yield strength. Because weld flaws are already formed, they do not require a stress threshold to nucleate. However, most of the weld flaws are embedded within the material and not initially exposed to the environment. Thus, such flaws will not propagate until exposed to the environment. As general corrosion proceeds, some initially embedded weld flaws may be exposed to the environment while others are corroded away.

All regions of the Alloy 22 waste package outer barrier, except the outer-closure lid welds, are solution-annealed before the waste packages are loaded with SNF assemblies. Thus, the waste package OCB will be very unlikely to develop residual stresses or stress intensity factors sufficiently high for SCC to occur provided that fabrication defects in the waste package OCB are not present that can lead to stress corrosion cracking as noted in Section 6.2.1. The mean value for the probability that a waste package OCB has at least one such defect was estimated using a Monte Carlo sampling process on the collective set of waste package OCB fabrication flaws (excluding OCB closure lid weld flaws) resulting in a value of 1.13×10^{-4} per waste package from Table 4.1-1. This probability is independent of time in the postclosure period since it arises from fabrication processes.

The outer closure lid weld is plasticity burnished to produce a layer of compressive stress that prevents SCC initiation until general corrosion removes this layer. The probability of having at least one undetected flaw in the outer closure lid weld has been estimated to be 0.156 from Table 4.1-1. However, such flaws are preferentially oriented in the circumferential direction along the weld, which will not propagate due to the residual hoop stresses. Only a small fraction (i.e., 0.008 from Table 4.1-1) of these flaws are likely to be oriented sufficiently normal to the residual hoop stress direction to permit propagation once the compressive stress layer is removed by corrosive action. Thus, the probability of a closure lid weld having a flaw that can propagate is given by $0.156 \times 0.008 = 1.25 \times 10^{-3}$. Specifications for controlled or low-plasticity burnishing of the waste package OCB closure lid call for the outer lid weld to be stress mitigated to a compressive depth of at least three mm (SNL 2007 [DIRS 179394], Table 4-1, Item 03-17). The upper values for the range of general corrosion rates for Alloy 22 from Section 4.1.14 is 15 nm/yr, which implies the compressive layer will survive for $\geq 2 \times 10^{5}$ years in the early failure event scenario provided the low-plasticity burnishing process is properly performed.

6.3.2 Screening Analysis for the Nominal (Early Failure) Event Scenarios

Initiating events where the OCB is potentially breached for this early failure scenario include the failure of the low-plasticity burnishing process such that the compressive stress layer in the waste package OCB closure lid is not produced, failure of the waste package OCB stress mitigation processes to function properly, weld flaws in the waste package OCB lid, and failure to properly emplace drip shields. Weld flaws in the waste package OCB lid or a failure of the stress mitigation processes can lead to a waste package breach from either weld flaw propagation or SCCs initiated by the residual stresses. Localized corrosion of a waste package OCB can occur if a suitable environment develops where the two primary components are elevated temperature and the presence of advective flow (Output DTN: MO0705CRITPROB.000, file: *DSLC 01–29–08.zip*). A drip shield emplacement error could result in an advective flow path to the waste package OCB creating an environment for subsequent localized corrosion processes that could breach the waste package OCB.

These events are analyzed in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007 [DIRS 178765], Section 6.3.5). If a flaw that is approximately normal to the circumferential tensile stress exists, SCC can occur since the weld flaw is an SCC initiator. The propagation rate for SCCs in Alloy 22 is given in Section 4.1.14 as 1.1×10^{-9} mm per second, which will penetrate the 25-mm-thick waste package OCB lid in < 1,000 years, causing a breach. Events requiring probability values for the screening calculation are listed as follows:

1. Probability of a failure for the low-plasticity burnishing process on the waste package OCB closure lid, or a failure of processes for stress mitigation for the waste package OCB, or a drip shield emplacement error

2. Probability of improper absorber material in a canister

3. Probability of a loading curve violation for a PWR TAD canister.

The probabilities of events in this scenario are derived from preclosure activities, making those values independent of the postclosure period. The mean value of the probability distribution for failure of the low-plasticity burnishing process is given in DTN: MO0705EARLYEND.000 [DIRS 180946], Table 1, as 3.84×10^{-5}. The probability that a waste package OCB closure weld has a flaw that can propagate through the OCB was estimated previously as 1.25×10^{-3}per waste package. The mean probability of waste package OCB fabrication defects as 1.13×10^{-4} per waste package and the mean probability value for improper emplacement of a drip shield is given in Table 4.1-1 as 4.36×10^{-9} per drip shield. The probability of localized corrosion breaching the waste package OCB from advective seepage flow resulting from a misplaced drip shield is conservatively set to 1.0. The probability of installing improper absorber plate material in a TAD or DOE canister is a fabrication related error. This type of error was evaluated in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007 [DIRS 178765], Section 6.3.2). The mean value of the probability distribution for a fabrication failure is given in DTN: MO0705EARLYEND.000 [DIRS 180946], Table 1, as 1.25×10^{-7} per canister.

An analysis of commercial SNF misload probabilities was documented in *Commercial Spent Nuclear Fuels Waste Package Misload Analysis* (BSC 2003 [DIRS 166316]). Results from this analysis establish that the probability of a loading curve violation in a 21-PWR Absorber Plate Waste Package is 1.18×10^{-5} (BSC 2003 [DIRS 166316], Table 41). The TAD canister specifications require the canisters for pressurized water reactor SNF to contain 21 assemblies similar to the 21-PWR Absorber Plate Waste Package (SNL 2007 [DIRS 179394], Section 4.1.1.2). The cited analysis is used as a surrogate for misloading waste forms in a TAD canister since the misloading of an assembly into a TAD canister requires the same improper selection of an assembly with characteristics (burnup and enrichment) in the unacceptable range of the loading curve. Thus, the probability of a loading curve violation for TAD canisters is expected to be similar in magnitude to the 21-PWR Absorber Plate Waste Package value. However, neighboring assemblies that have low reactivity values may provide partial compensation for the excess reactivity from the incorrectly loaded assembly. Given that a misloading curve violation occurs, the likelihood of the misloaded configuration having potential for criticality has been shown to be 0.014 from results of a probabilistic calculation of that potential (SNL 2008 [DIRS 182788], Section 7).

The probability of misloading assemblies in the 44-BWR TAD canister is insignificant since the entire expected BWR inventory for the repository is in the acceptable region of the loading curve map (SNL 2008 [DIRS 182788], Section 6.1.1.1.3). Misloading of waste forms in DOE-owned SNF canisters is very improbable because the shape and size of the DHLW glass canisters and the various DOE-owned SNF canisters differ significantly and can be readily distinguished by visual inspection per Section 4.1.5. Thus, the waste form misload probability for DOE-owned SNF waste packages is considered sufficiently low such that, if quantified, would not significantly increase the overall probability of criticality in the repository.

Sensitivity studies have shown that the pressurized water reactor SNF waste form in various degraded configurations such as saturated porous schoepite does not result in a more reactive configuration than the design basis configuration (SNL 2007 [DIRS 181373], Table A-12; SNL 2008 [DIRS 182788], Section 6.2.5). This result supports the assertion (Section 1) that a loading curve violation is the most likely pressurized water reactor waste form configuration with potential for criticality. The probability of a potentially critical configuration resulting from an assembly misload of a PWR TAD canister, from the above discussion, is $0.014 \times 1.18 \times 10^{-5} = 1.65 \times 10^{-7}$ per TAD canister.

The probability for the occurrence of configurations with potential for criticality is evaluated from a number of independent sets of sequences of events where all of the events in any specific sequence must happen for that configuration to occur. Since the events in any one sequence can also be considered as independent entities, the probability of the sequence is the product of the probability of each individual event. The expected probability of having a particular sequence occur in exactly k waste packages in the repository is a Binomial process described by the Binomial probability distribution, $P_B(n; p, N)$, with probability "p" for occurrence in a waste package and "q = 1 - p" for non-occurrence. The probability of having the sequence occur in at least "$k+1$" waste packages is given by:

$$P(k+1 \text{ or more items occur}) = 1 - P_B(k; p, N) \qquad \text{(Eq. 6.3-1)}$$

where

k = number of items affected (e.g., waste packages, drip shields)
p = probability for occurrence of the event
N = number of possible items involved.

For large N and small "p" where $N \times p \cong \lambda$, the Binomial distribution converges to the Poisson distribution with a mean of $\lambda = N \times p$. Then Equation 6.3-1 can be written as:

$$P(k+1 \; or \; more \; waste \; packages) = 1 - P_P(k; N \times p) \cong 1 - \frac{\lambda^k \times \exp(-\lambda)}{k!} \qquad \text{(Eq. 6.3-2)}$$

The criterion for screening criticality scenarios from consideration in the repository is having a low probability for the occurrence of a criticality event sequence for any waste package in the repository (which can be stated as the probability of having at least one such sequence occur) is given by Equation 6.3-2 with $k = 0$. For the case where $k = 0$ and λ is small, Equation 6.3-2 can be approximated by λ. Then the probability of at least one waste package configuration with criticality potential occurring in the repository is given by λ ($= N \times p$).

The initiating event leading to a possible waste package early failure scenario is a SCC caused breach of the waste package OCB. Initiators for SCCs, discussed above, are OCB closure lid weld flaws having a per package probability of $3.84 \times 10^{-5} \times 1.25 \times 10^{-3}$, OCB fabrication flaws having a per package probability of 1.13×10^{-4}, and a misplaced drip shield coupled with localized corrosion having a per package probability of $4.36 \times 10^{-9} \times 1.0$, where the probability of localized corrosion is set to 1.0. The combined probability of the initiators for the suite of early failure scenario evaluations is given by:

$$(3.84 \times 10^{-5} \times 1.25 \times 10^{-3}) + (1.13 \times 10^{-4}) + (4.36 \times 10^{-9} \times 1.0) = 1.13 \times 10^{-4}$$

Evaluating the event sequences for commercial SNF and DOE-owned SNF with potential for criticality using the number of 21-PWR TAD canisters given in Table 4.1-3 as 4,568, the number of 44-BWR canisters as 2,915, and DOE-owned SNF canisters with criticality potential (DOE1, DOE2, and DOE7 groups) as 1,223 and setting the number of drip shields equal to the number of waste packages gives:

PWR TAD canister loading curve violation:
$\{1\text{-}P_B (0; ((3.84 \times 10^{-5} \times 1.25 \times 10^{-3} + 1.13 \times 10^{-4} + 4.36 \times 10^{-9} \times 1.0) \times$
$1.65 \times 10^{-7}), 4568)\} = 8.5 \times 10^{-8}$

PWR TAD canister absorber misload:
$\{1\text{-}P_B (0; ((3.84 \times 10^{-5} \times 1.25 \times 10^{-3} + 1.13 \times 10^{-4} + 4.36 \times 10^{-9} \times 1.0) \times$
$1.25 \times 10^{-7}), 4568)\} = 6.5 \times 10^{-8}$

44-BWR TAD canister absorber misload:
$\{1\text{-}P_B (0; ((3.84 \times 10^{-5} \times 1.25 \times 10^{-3} + 1.13 \times 10^{-4} + 4.36 \times 10^{-9} \times 1.0) \times$
$1.25 \times 10^{-7}), 2915)\} = 4.1 \times 10^{-8}$

DOE-owned SNF canister absorber misload (DOE1, DOE2, and DOE7):
$\{1\text{-}P_B (0; ((3.84 \times 10^{-5} \times 1.25 \times 10^{-3} + 1.13 \times 10^{-4} + 4.36 \times 10^{-9} \times 1.0) \times$
$1.25 \times 10^{-7}), 1223)\} = 1.7 \times 10^{-8}$

Evaluating the event sequences for DOE-owned SNF with the additional absorber loading constraint from Section 4.1.15 that the DOE1 waste form (MOX) and DOE7 waste form (aluminum-based DOE-owned SNF) include neutron absorber shot as well as plate type absorbers that eliminates these waste forms from the set that has potential for criticality results in an estimated DOE-owned SNF canister absorber misload probability given by:

DOE-owned SNF canister absorber misload (89 DOE2 canisters, Table 4.1-2):
$\{1\text{-}P_B\ (0; ((3.84 \times 10^{-5} \times 1.25 \times 10^{-3} + 1.13 \times 10^{-4} + 4.36 \times 10^{-9} \times 1.0) \times$
$1.25 \times 10^{-7}), 89)\} = 1.3 \times 10^{-9}$.

Thus, a conservative estimate for the probability of achieving a configuration with criticality potential in the repository due to early failure initiating events, based on summing the results above, including the DOE1, DOE2, and DOE7 contributions is 2.1×10^{-7} for 10,000 years. The estimate, including only the DOE2 contribution, is 1.9×10^{-7} for 10,000 years. Since the events in the above evaluation are all associated with operations during the preclosure period, the probabilities are constant over the postclosure time period.

The critical configuration is potentially achievable for waste packages for which either an absorber misload or loading curve violation has occurred in combination with either: 1) a schoepite-moderated system as a result of water vapor entry through SCC or 2) a flooded configuration resulting from drip shield breach and subsequent OCB failure due to localized corrosion. These probability evaluations have been developed for the in-package degraded scenario, FEP 2.1.14.16.0A (Table 1.2-1). The events in the in-package intact configuration scenario, FEP 2.1.14.15.0A, are the same as those for the in-package degraded scenario and do not increase the probability of achieving a configuration with potential for criticality. The probability values for FEP 2.1.14.15.0A are thus insignificant.

An early failure induced breach of a waste package is not expected to increase the criticality potential for the near-field or for the far-field configurations (FEPs 2.1.14.17.0A and 2.2.14.09.0A, respectively) since the waste package breach is limited to SCCs that, with the exception of a drip shield misplacement event, do not permit sufficient accumulation for criticality in the external environment. The probability of drip shield misplacement followed by localized corrosion is sufficiently low (4.36×10^{-9} per drip shield $\times$ 10,557 $\times$ P(LC) < 4.6×10^{-5}, where P(LC) is the probability of localized corrosion) to limit the contribution from this initiating event to an insignificant level. A discussion of the events required for external critical configurations is provided in Section 4.1.15 with the conclusion that the likelihood for the occurrence of configurations with potential for criticality was very low. Thus, the criticality potential in the near-field and far-field locations referenced by FEPs 2.1.14.17.0A and 2.2.14.09.0A from an early failure event that breaches the waste package is insignificant.

Events such as the following have probabilities less than one, some much less than one and, individually or in combination, impact the probability that the early failure event scenario can lead to configurations that have criticality potential. This list is not exhaustive nor is the supporting discussion for each complete. However, it illustrates some of the additional events in the early failure event scenario for which information is unavailable to adequately quantify the probability, but would be necessary for achieving configurations that have criticality potential.

- Accumulation or presence of a critical mass of fissionable material—Credit is not taken for the stainless steel liner or TAD canister as a barrier for commercial SNF waste packages that are breached, and loss of the barrier capability of the cladding must also be assumed since proving otherwise is difficult if not impossible due to the complexity of the event. Thus, schoepite will likely form from the fissile material. The likelihood that the resultant distribution of fissile material would rearrange into a near optimum arrangement conducive to criticality is low since the structures are intact at the initiation of a breach. Waste package lid failure is expected to be limited to SCCs, allowing gas exchange including water vapor with the drift but preventing sizeable objects from escaping. Thus, although commercial SNF rearrangement is hypothetically possible within the assembly containment tubes, absorber plates are required to extend the full length of the assemblies that limit the effect of fissile material rearrangements on reactivity. Likewise, no credit is taken for the stainless steel liner or the DOE canister as a barrier for codisposal waste packages that are breached with similar arguments concerning the likelihood of accumulation and/or rearrangement of the fissile material.

- Separation of fissionable material from the neutron absorber material—The waste forms and absorber materials for commercial SNF and codisposal waste packages are expected to remain inside the waste packages after becoming breached as discussed previously. After seepage water has returned, there is little possibility of moving much of the chromium boride particles from the vicinity of the spent fuel. *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007 [DIRS 181165], Section 6.3.3) indicates that, due to the boron in borated stainless steel having a very low solubility within the iron matrix of the steel, the boron is present as separate chromium boride particles instead of a solid solution. These particles do not dissolve into the aqueous solution during degradation of the steel but are left behind as insoluble products during corrosion. Therefore, the neutron absorber for commercial SNF waste packages is expected to remain between fuel cell regions, but some degraded configurations of the waste form may decrease the effectiveness of the absorber. This mechanism is unlikely to result in a critical configuration since the corrosion rate of the Ni-Gd neutron absorber material is very low (Table 4.1-8) with < 6 mm (likely much less) removed over the 10,000-year period. In addition, the gadolinium in the DOE-owned SNF canisters forms phosphate or carbonate corrosion products (SNL 2007 [DIRS 181165], Section 6.3.16), which have very low corrosion rates.

- Presence of a moderator—Seepage rates and the likelihood of the either commercial SNF or codisposal waste packages experiencing seepage.

After the cladding is breached within a failed waste package, the interior of the fuel rod will be exposed to the repository environment. As the temperature of the repository decreases after the initial thermal spike, the relative humidity to which the commercial SNF matrix will be exposed will increase and is expected to approach 100% when the temperature decreases to 100°C and lower (BSC 2004 [DIRS 169987], Section 6.2.2.2). The plausible mechanisms for waste form degradation are discussed in detail in *CSNF Waste Form Degradation: Summary Abstraction* (BSC 2004 [DIRS 169987], Section 6.2.2), but these discussions indicate that the overall oxidative dissolution process involves a coupled series of redox, surface complexation and dissolution, and precipitation reactions depending on the fluid environment (water film on the fuel surfaces). Upon contact with air-saturated condensate (i.e., water), UO_2 (and commercial

SNF) is expected to undergo reactions of the following type to form dehydrated schoepite and metaschoepite (this molecule is referred to as schoepite throughout this report):

$$UO_2\ (s) + \tfrac{1}{2}\ O_2\ (aq) + 0.8\ H_2O\ (l) = UO_3{\cdot}0.8H_2O\ (s)$$

$$UO_2\ (s) + \tfrac{1}{2}\ O_2\ (aq) + 2\ H_2O\ (l) = UO_3{\cdot}2H_2O\ (s)$$

Carbon and silica are much less effective moderators than water, and their introduction into commercial SNF waste packages from seepage infiltration will displace water and effectively reduce the k_{eff} of the system, thus reducing the potential for criticality. Additionally, carbon and silica can act as a neutron reflector. However, inside the waste package, their reflector effects, which increase the k_{eff}, are secondary to their water-displacement effects, which decrease the k_{eff} of the system. Water, the most effective neutron-moderating material, can enter the waste package as advective flow through breaches resulting from localized corrosion or as vapor diffusion through SCCs. The total volume of water that can enter through SCCs over 10,000 years is insufficient to support criticality prior to degradation as discussed in *Waste Package Flooding Probability Due to Seismic Fault Displacement* (SNL 2008 [DIRS 184078], Section 6.2.1.4). Localized corrosion associated with advective flow onto waste packages through misplaced drip shields could provide a means for advective to enter the waste packages. However, this latter mechanism is expected to be very unlikely since the proper environment (temperature and chemistry) must be present to initiate localized corrosion. The principal mechanism for water retention in a waste package would be through reactions of the oxidized fuel with humid air resulting in the precipitation of the schoepite mineral. Not all of the uranium atoms are expected to form into fully hydrated schoepite. This is because as schoepite forms, it starts at the exposed surface of the fuel pellets/fragments and moves inwards towards the center of the pellet/fragment depending on the availability of oxygen and hydrogen. This is a diffusion controlled process as the oxidizing and hydration molecules must diffuse through the material already formed, which will effectively limit the oxidation rate for the inner uranium atoms, slowing the rate of schoepite formation. For the early failure scenario, the waste package configuration with fuel tubes intact is expected to be maintained. Thus, any schoepite that possibly separates from the pellets will remain in the fuel tube and act as resistance to the diffusion process as the initial void volume in the fuel tube decreases.

Given an initiating event, conditions inherent in the use of a truncated sequence of events to estimate a conservative value for the probability of achieving a configuration with potential for criticality were discussed in this section. Corroborating evidence supporting this assertion discussed earlier indicates that the resultant probability of criticality from events associated with the early failure event scenario would likely be significantly smaller than the estimated upper bound.

6.4 FEPS ASSOCIATED WITH SEISMIC EVENT SEQUENCE INITIATORS

Seismic disruptive event criticality FEPs identify scenarios that may initiate a sequence of events resulting from seismic activity in various repository locations that could potentially lead to a critical event. These scenarios cover the in-package environment, near-field environment, and far-field environment. This section evaluates the criticality potential resulting from events initiated by seismically induced ground motion. Vibratory ground motion, faulting, and rockfall induced by a seismic event are potential initiating events that could cause drip shield damage through SCCs and/or rupture leading to subsequent waste package damage. Such events may allow the influx of water (either advective or diffusive) into the waste package, which, in turn, has the potential to initiate processes leading to a criticality.

A seismic event can induce fault displacements that can potentially lead to drip shield and waste package failure for those structures intersecting the fault. While one requirement for subsurface development is that a minimum standoff distance of 60 m be maintained between Quaternary faults with potential for significant displacement and repository placement drifts (SNL 2007 [DIRS 179466], Table 4-1, Item 01-05), uncertainty exists concerning the location of such faults at the repository horizon (SNL 2007 [DIRS 176828], Section 6.11.2.2). Given a fault-generated failure, advective or diffusive flow can potentially enter the waste package, leading to conditions conducive to criticality. Seismic events that can cause significant displacement (> 0.1 cm) along fault lines that do intersect the drifts have a low probability of occurrence (i.e., mean annual exceedance frequencies of less than 10^{-6} per year) (SNL 2007 [DIRS 176828], Table 6-61). Additionally, new fractures that intersect the drift segments and the collapsing of the drift due to a seismic event will have an effect on the seepage with respect to both location and rate. However, these changes in seepage have no impact on the repository's potential for criticality without drip shield failure.

Seismically induced ground motion causing failure of the host rock has the potential to disrupt the integrity of EBS components and, in particular, the waste packages, which could potentially lead to breaching of the waste package. Seismically induced deformation of EBS components may result in plastic yielding or even failure of EBS components. If the residual stress on a plastically deformed EBS component exceeds a threshold value, then accelerated stress corrosion cracking is inferred to occur that may result in the formation of transport pathways for seepage (SNL 2007 [DIRS 178851], Section 6.2). The area in which the residual stress threshold is exceeded is conceptualized to result in the formation of a tightly spaced network of stress corrosion cracks (SNL 2007 [DIRS 178851], Section 6.2.1).

The seismic disruptive event criticality FEP scenarios are identified as FEPs 2.1.14.18.0A, 2.1.14.19.0A, 2.1.14.20.0A, and 2.2.14.10.0A (Table 1.2-1). The scenarios associated with FEPs are: (1) in-package criticality resulting from a seismic event (intact configuration), (2) in-package criticality resulting from a seismic event (degraded configuration), (3) near-field criticality resulting from a seismic event, and (4) far-field criticality resulting from a seismic event. The criticality potential of intact configuration scenario is negligible since intact configurations will remain subcritical since any degradation of intact configurations is insufficient by design to compromise the functioning of criticality control structures (Section 1.2). Although configurations not conforming to design specifications are applicable to both intact and degraded scenarios, configurations with potential for criticality require sufficient water for moderation. These latter configurations (i.e., sufficient moderation) can only occur in degraded scenarios (Section 6.3) and are considered in those analyses. The criticality potential of configurations for the degraded in-package scenario is dependent upon the probability of a seismic event breaching a waste package in combination with other events, notably misload configurations. If neutron absorber material is present and if the PWR TAD canisters are loaded according to the applicable loading curve, the criticality potential of the waste package is insignificant; otherwise, there may be a potential for criticality provided adequate moderating material is present along with other factors.

6.4.1 Waste Package Failure from a Seismic Event

Analyses of damage to drip shields from fault displacements expected to be applicable to the first 10,000-year-period after repository closure considers the drip shields to be intact prior to the event for determining the clearances between EBS components (SNL 2007 [DIRS 176828], Section 6.11.1). Damage to the drip shield causing loss of function is not expected to result from seismic faulting until sufficient displacement occurs to make contact between the drip shield and the drift. The emplacement drift has a nominal diameter of 5.5 m (5,500 mm) (SNL 2007 [DIRS 179466], Table 4-1, Item 01-10). Within the drift, the steel support beams and associated ballast form a level invert with a surface height of 52 in (1,320.8 mm) above the lowest part of the drift (SNL 2007 [DIRS 179354], Figure 4-1). The drip shield is a free-standing structure that sits on the invert. The drip shield has an external height for the overlap section of 113.62 in (2,886 mm) (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), rounded up to 2,890 mm. The internal height of the drip shield, defined as the distance from the invert floor to the lowest point on the underside of the top of the drip shield, is 106.93 in (2,715.8 mm) (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), which is rounded to 107 in (2,717.8 mm). The clearance between the crown (top) of the drip shield and the drift roof is 50 in (SNL 2007 [DIRS 179354], Table 4-2, Item 07-01), rounded to the nearest inch or 1,270 mm.

Seismic faulting can generate a large number of possible dynamic response scenarios in a drift. A reasonable approach for simplifying the analyses was to calculate clearances excluding the pallet elevations[5] (SNL 2007 [DIRS 176828], Section 6.11.1.1). The combined clearance between the crown of the drip shield and the roof of the drift (1,270 mm) and between the top of the waste package and the bottom of the drip shield, as shown in Table 6.4-1, determines the maximum fault displacement that could occur before the waste packages are potentially damaged or breached through a shearing mechanism. This analysis selected the smaller of these clearances since drift collapse is likely in the lithophysal zone during a seismic disruptive event associated with fault displacement. Fault displacement in excess of the clearance values in Table 6.4-1 are conservatively considered to fail the waste package and the overlying drip shield.

The set of clearance values in Table 6.4-1 represents the failure criterion for waste packages and drip shields under fault displacement when the waste package OCB and drip shield are intact at

[5] The following is the rationale for neglecting the pallet elevation (SNL 2007 [DIRS 176828], Section 6.11.1.1):

> Movement along a sudden discontinuity will affect the rubble surrounding the drip shield after drift collapse in the lithophysal zone. The lithophysal rubble is a loosely packed material with porosities between 0.09 and 0.29. (The porosity of rockfall in the nonlithophysal units is similar to that for the lithophysal rubble.) With this free space, the rubble has substantial movement in the plane of discontinuity and longitudinally along the drift axis during the fault displacement. The movement of the rubble will allow the drip shield to move with the fault displacement, rather than being rigidly pinned to the invert. In this situation, the effective clearance around the drip shield is expected to be significantly larger than space between the top of the waste package and bottom of the drip shield.
>
> Simulations demonstrate that the rubble particles undergo large dynamic motion in response to displacements of the drip shield, similar to what would occur during a vertical fault displacement. It follows that the clearance between the top of the drip shield and the roof of the drift will be partly available, but the exact value is difficult to quantify. Likewise, the dynamic response of the rubble, invert and emplacement pallet during a fault displacement is difficult to predict. As a simplification, the approximation is made that the clearance between the top of the waste package and the bottom of the drip shield is determined without the pallet. This is a reasonable approximation because the clearance between the top of the drip shield and the roof of the drift, 1,270 mm, is more than four times greater than the differences in clearance with or without considering the pallet, which range from 283 to 317 mm (SNL 2007 [DIRS 176828], Table 6-57).

the time of the seismic event. These clearances are appropriate during the first 10,000 years after repository closure. The clearances in Table 6.4-1 exclude the pallet elevation and no credit is taken for the downward collapse of the invert.

Table 6.4-1. Maximum Allowable Displacement with Drift Collapse for an Intact Drip Shield

Package Type	Outside Diameter of OCB (mm)	Nominal Length (mm)	Clearance Without Pallet (mm)
Commercial SNF TAD Canister	1,881.6	5,850.1	836
Codisposal Short	2,044.7	3,697.4	673
Codisposal Long	2,044.7	5,303.9	673
Codisposal MCO	1,749.3	5,278.6	969

Source: Table 4.1-3.

NOTES: Clearance without the pallet is calculated as the interior height of the drip shield (2,717.8 mm) minus the outside diameter of the waste package OCB, rounded to three significant digits.

MCO = multicanister overpack, OCB = outer corrosion barrier.

The number of waste packages that could be emplaced on faults in the repository is evaluated following the analysis method from DTN: MO0705FAULTABS.000 [DIRS 183150], file: *Fault Displacement Abstraction for Criticality.xls*, spreadsheet: "Tables by WP Type," adjusted for the inventory from Table 4.1-2 and dimensions from Table 6.4-1. Table 6.4-2 lists the expected number of waste packages by type that could be emplaced on fault lines in the repository. Tables 6.4-4 and 6.4-5 show the result, developed in Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," of combining the exceedance frequencies that cause failure and the number of packages potentially emplaced on faults from Table 6.4-2. The total number of all waste packages impacted by each fault is calculated from the number of fault intersections with the emplacement drifts. The expected number of waste packages for each of the waste package groups on a particular fault is estimated based on the percent of total length of waste packages. The potential for waste package damage from a seismic event is a function, among other variables, of the exceedance frequency of the event and waste package clearance distance. Thus, the damage potential of a seismic event varies with the particular fault. The fractional length proportioned to each waste package type is used to determine the cumulative number of waste packages expected to fail by type as a function of annual exceedance frequency. These variations result in a range of exceedance frequencies for the various waste package variants as shown in Tables 6.4-4 and 6.4-5.

Table 6.4-2. Expected Number of Waste Packages by Type Emplaced on Faults

Fault	Commercial SNF TAD Canister	Codisposal Short	Codisposal Long	Codisposal MCO
3 - Drill Hole Wash, Pagany Wash, and Sever Wash	19.4	2.6	3.5	0.5
4 – West Ghost Dance	8.2	1.1	1.5	0.2
5 – Sundance	4.5	0.6	0.8	0.1

Table 6.4-2. Expected Number of Waste Packages by Type Emplaced on Faults (Continued)

Fault	Commercial SNF TAD Canister	Codisposal Short	Codisposal Long	Codisposal MCO
Sites 7a/8a	127.7	17.3	22.8	3.2
Totals	159.8	21.6	28.5	4.0

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: “Tables by WP Type,” rows 171 to 181.

MCO = multicanister overpack.

Table 6.4-3. Cumulative Number of Failed Commercial SNF Waste Packages Expected versus Annual Exceedance Frequency

Exceedance Frequency Range (1/yr)	Commercial SNF TAD Canister
$> 8.2 \times 10^{-8}$	0
7.0×10^{-8} to 8.2×10^{-8}	19.4
2.7×10^{-8} to 7.0×10^{-8}	27.6
1.0×10^{-8} to 2.7×10^{-8}	32.1

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: “Tables by WP Type,” rows 183 to 191.

Table 6.4-4. Cumulative Number of Failed Codisposal Waste Packages Expected versus Annual Exceedance Frequency

Exceedance Frequency Range (1/yr)	Expected Number of Failures Codisposal Short	Expected Number of Failures Codisposal Long	Exceedance Frequency Range (1/yr)	Expected Number of Failures Codisposal MCO
$> 1.2 \times 10^{-7}$	0	0	$> 6.3 \times 10^{-8}$	0
1.1×10^{-7} to 1.2×10^{-7}	2.6	3.5	5.4×10^{-8} to 6.3×10^{-8}	0.5
4.1×10^{-8} to 1.1×10^{-7}	3.7	4.9	2.1×10^{-8} to 5.4×10^{-8}	0.7
1.3×10^{-8} to 4.1×10^{-8}	4.3	5.7	1.0×10^{-8} to 2.1×10^{-8}	0.8
1.0×10^{-8} to 1.3×10^{-8}	21.6	28.5		

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: “Tables by WP Type,” rows 183 to 192.

MCO = multicanister overpack.

6.4.2 Consequences of Seismic Vibratory Ground Motion for Waste Packages and Drip Shields

The effects of vibratory ground motion depend on the condition of the components and the in–drift environment (e.g., an early seismic event after repository closure) may not significantly damage components but a second event of the same magnitude at some later time may cause significant damage if the components have been weakened by general corrosion (SNL 2007 [DIRS 176828], Section 6.1.2). The predominant mechanism for damage is seismically induced impact between EBS components. Under significant vibratory ground motions, impacts may occur between adjacent waste packages, between a waste package and its emplacement pallet, and between waste packages and the surrounding drip shield. It was concluded that most of the damage to waste packages would be caused by the waste package-to-pallet impacts. However damage such as rupture (e.g., tearing) from vibratory ground motion is expected to occur only for degraded waste packages subjected to large ground motions (SNL 2007 [DIRS 178851], Section 8).

The impacts on drip shields and waste packages from vibratory ground motions is documented in *Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion* (SNL 2007 [DIRS 178851], Section 8) and in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6). The major conclusions from those assessments are:

- TAD canister waste packages were quite robust with respect to vibratory motions even with allowance for 2 mm of general corrosion thinning (SNL 2007 [DIRS 178851], Section 6.5.1.2.1) to the outer corrosion barrier
- A major portion of damage to the waste package outer corrosion barrier was the result of waste package-pallet impacts
- Failure of the TAD waste package OCB is expected for only a limited number of waste packages at high peak ground velocity values
- Codisposal waste packages are less robust than the TAD waste packages and the waste package OCBs are susceptible to seismic induced impact damage
- Rockfall impact or rubble loading of the drip shields may cause plate rupture allowing advective seepage flow to contact the waste packages in both lithophysal and nonlithophysal zones
- Analyses of large, single-block (28.29 metric tons) impacts of rocks in the nonlithophysal zone show that such impacts may cause the drip shield to buckle and potentially contact the waste package outer corrosion barrier.

Impacts by a single rock block having a mass less than 28.29 metric tons (last bullet) can cause drip shield plate rupture but are not predicted to cause the drip shield to fail as a barrier to rockfall nor to contact the waste package, thus the waste package is not damaged by such impacts (SNL 2007 [DIRS 178851], Section 6.4.7.3). It is therefore reasonable to screen out damage to the waste package from rockfall in the nonlithophysal units (SNL 2007 [DIRS 176828], Section 6.10.2.11) with the exception of impacts from very large rock masses (≥ 28.29 metric tons). The effects of these types of rock impacts are discussed in more detail in Section 6.5.

Stress corrosion cracking from high residual stress is expected to be the cause of waste package damage from impact processes under vibratory ground motion (SNL 2007 [DIRS 178851], Section 8.2). Regions where the residual stress from mechanical damage exceeds the tensile failure criterion are expected to be severely cold-worked and, hence, potentially subject to enhanced SCC. However, if cracking were to occur as a result of specific environmental conditions coincident with the mechanical deformation, cracks would take time to develop after the shaking event causes a change in loading.

A rupture or tear may occur in a drip shield plate if the local strain exceeds the ultimate tensile strain due to either loading of the drip shield from drift collapse in the lithophysal zone or rock block impacts in the nonlithophysal zone on the drip shield caused by vibratory ground motion (SNL 2007 [DIRS 179476], FEP 2.1.03.03.0C, FEP 2.1.03.03.0B). Localized corrosion could potentially cause waste package failure from exposure to advective seepage flow following rupture of a drip shield (SNL 2007 [DIRS 176828], Section 6.1.4). The most likely form of localized corrosion to affect the waste package OCB is crevice corrosion (Section 4.1.14) that can attack discrete locations such as occluded regions where contact exists between the waste package OCB and pallet if environmental conditions favorable to the corrosion processes are present. A condition necessary for localized corrosion is the persistent presence of an aqueous medium on the waste package OCB surface and dissolved chemical ions. Environmental conditions conducive to localized corrosion are present only for portions of the initial 10,000-year period following repository closure. Localized corrosion, once initiated, can penetrate the waste package OCB in less than 1,000 years (Section 4.1.14).

For a criticality event to occur, the proper combination of materials (neutron moderators, neutron absorbers, fissile materials, or isotopes) and geometric configuration must exist as stated in Section 6.2. The presence of neutron absorber materials in waste package canisters is important for criticality control during the 10,000-year period following repository closure for the majority of the canisters proposed for disposal of SNF in the repository. For seismically induced vibratory events, there are no mechanisms identified that can lead to removal of neutron absorber material from a waste package. For such a situation to occur, vibratory ground motions would need to induce failure of the spent fuel canisters and fracture the fuel baskets. In addition, there are no forces identified that can systematically separate the absorber materials from the fuel material that would result in concentrating absorber in one part of a waste package and fuel in another. It has been previously demonstrated through loading curve analyses for the 21–PWR Absorber Plate and the 44–BWR Absorber Plate waste package variants that an intact, fully flooded waste package configuration as designed cannot achieve criticality (SNL 2008 [DIRS 182788], Section 6.2.2). The 21-PWR and 44-BWR TAD canisters are similarly expected to not have any criticality potential in an aqueous environment provided the borated stainless steel (or its degraded form as boron carbide) absorber proposed for use in the TAD canisters remains in the proximity of the waste form. However, neutron absorber material misloads can occur in TAD canisters as the result of various operations (or the lack thereof) during the canister fabrication and loading processes. These processes include the use of wrong materials and/or failure to install the specified neutron absorber materials into the canister, which affect the criticality potential of waste forms. Likewise, the DOE-owned SNF canisters do not have any criticality potential in an aqueous environment provided the Ni–Gd absorber proposed for use in the DOE-owned SNF canisters remains in the proximity of the waste form (BSC 2006 [DIRS 181335], Section 7.10) and Section 4.1.15.

Vibratory motions from seismic events could theoretically cause schoepite in a breeched commercial SNF waste package to migrate to the ends of the fuel assembly tubes, fall to the bottom of a waste package, and thus separate from the neutron absorber material. This scenario is expected to be very improbable for reasons that include the following:

- The fuel assembly tubes extend the full length of the fuel assemblies allowing minimal clearance for material to pass

- The active fuel is centered in the fuel assemblies away from the ends where losses could occur
- Assemblies will likely have their end caps attached when loaded into the TAD canisters
- Assembly hardware (e.g., spacer grids and end fittings) will limit the magnitude of any lateral movement
- The clearance between the fuel assembly tubes and end plates is ≤ 1 inch.

Thus, for commercial SNF, the fissile material will likely remain within the fuel tubes that contain the neutron absorber material, minimizing the likelihood that a critical configuration could assemble at the bottom of a waste package from schoepite exiting the fuel assembly tubes.

The same scenario (separation of fissile and absorber material) does not exist for DOE-owned SNF since the DOE canisters have a different geometry (small (18-inch diameter, sealed container) and the canister must breach for material to leave the canister. Analyses have shown (Radulescu et al. 2004 [DIRS 165482], Section 11.4) that all intact and degraded configurations of the DOE-owned SNF have a k_{eff} below the critical limit provided the neutron absorber material is present as required. Thus, vibratory motions from seismic events that do not cause failure of the DOE canister have little likelihood of initiating events that could lead to configurations with potential for criticality.

As discussed in Section 6.3.1, the rate of water transport through cracks in both the drip shield and waste package under seepage drips indicates that the maximum volumetric flow rate through SCCs in a given waste package is very low (DTN: SN0705WFLOWSCC.001 [DIRS 184848], file: *Analysis for Water Flow through Stress Corrosion Cracking (SCC) Cracks in Waste Package and Drip Shield.xls*, spreadsheets: “Sheet flow DS flow rate” and “Sheet flow WP flow rate”) and insufficient to support localized corrosion. In addition, the interior of the waste packages will be warmer than the external environment for a considerable time period (SNL 2008 [DIRS 179962], Figure 6.4.2-4b). The accumulation of water within the waste package will be limited by evaporation through the breeches because of the warmer waste package as discussed in *Waste Package Flooding Probability Due to Seismic Fault Displacement* (SNL 2008 [DIRS 184078], Section 6.2.1.4). In addition, the intact configuration is designed to remain subcritical when fully flooded (SNL 2008 [DIRS 182788], Figure 6-6) and the design basis configuration accounts for corrosion loss of the neutron absorber over the first 10,000-year period following repository closure. However, it is very improbable that the cladding can maintain its barrier function during a vibratory ground motion event that damages the waste package OCB allowing schoepite to form after a breach develops. Thus, the set of events evaluated in Section 6.3.1 as contributors to the probability of criticality are appropriate for vibratory ground motions.

The series of events begins with the occurrence of a seismic vibratory ground motion event. Events in the various seismic vibratory scenarios requiring probability values for the calculation are listed as follows:

1. Probability of a seismic vibratory ground motion event

2. Probability of waste package OCB damage from effects of the ground motion

3. Probability of drip shield damage

4. Probability of improper absorber material in a commercial SNF or DOE-owned SNF canister

5. Probability of a loading curve violation for a PWR TAD canister.

6.4.2.1 Evaluation of EBS-Waste Package Impacts from Seismic Vibratory Ground Motion Events

For seismic events causing waste package-pallet impacts that can damage a commercial SNF waste package at the 90% residual stress level, the probability of damage is zero at a PGV value of 2.44 m/s (exceedance frequency of 4.518×10^{-7} per year). At a PGV value of 4.07 (exceedance frequency of 10^{-8} per year), the probability of impact damage is 0.118 (DTN: MO0703PASDSTAT.001 [DIRS 183148], file: *Kinematic Damage Abstraction 23-mm Intact.xls*, spreadsheet: "Probability of Damage"). Seismic events with the range of annual exceedance frequencies that can damage a TAD waste package are represented in the column labeled "PGV Value" in Table 6.4-5. The probability of a seismic event is a random event in time following a Poisson distribution (SNL 2007 [DIRS 176828], Section 5.2), which increases linearly in log-time. Thus, the probability that one or more of these events occur (i.e., one minus the probability that none occurs) is determined with Equation 6.3-1 and the information provided in Table 6.4-5:

Table 6.4-5. Probability of Seismic Vibratory Ground Motion Events Causing Damage to TAD Waste Packages

TAD Waste Package Variants					
PGV Value (m/s)	λ_1 (events/year)	λ_2 (events/year)	t_1 (years)	t_2 (years)	Probability
< 2.44	4.52×10^{-7}	NA	NA	NA	NA
2.44 to 4.07	1.0×10^{-8}	4.52×10^{-7}	10,000	0	4.41×10^{-3}

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," rows 239 to 244.

NA = not available.

The probability over 10,000 years for occurrence of a seismically induced vibratory ground motion event that could result in damage to the outer corrosion barrier of a TAD waste package from pallet-waste package impacts is 4.41×10^{-3} (Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type").

Seismic events causing waste package-pallet impacts that can damage a codisposal waste package are shown in Table 6.4-6 using the information from Table 6.4-6. The range of the seismic events are shown in the column labeled "PGV Value" with the associated annual exceedance frequencies in columns 2 and 3.

Table 6.4-6. Probability of Seismic Vibratory Ground Motion Events Causing Damage to Codisposal Waste Packages

PGV Value (m/s)	λ_1 (events/year)	λ_2 (events/year)	t_1 (years)	t_2 (years)	Probability
< 0.364	1.27×10^{-4}	NA	NA	NA	NA
0.364 to 0.4	9.30×10^{-5}	1.27×10^{-4}	10,000	0	2.87×10^{-1}
0.4 to 1.05	9.96×10^{-6}	9.30×10^{-5}	10,000	0	5.64×10^{-1}
1.05 to 2.44	4.52×10^{-7}	9.96×10^{-6}	10,000	0	9.07×10^{-2}
2.44 to 4.07	1.0×10^{-8}	4.52×10^{-7}	10,000	0	4.41×10^{-3}

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," rows 250 to 260.

NA = not available.

The probability over 10,000 years of a seismically induced vibratory ground motion event that could result in damage to the outer corrosion barrier of a codisposal waste package varies with the exceedance frequency range as listed in the last column of Table 6.4-6 (Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type").

If a seismic vibratory ground motion event occurs, the estimated probability of damage to a TAD waste package from impacts is given as 0.118 at the 4.07 m/s PGV range (Section 4.1.13), assuming a damage threshold at the 90% RST level, resulting in a probability of damage for a TAD waste package given by $4.41 \times 10^{-3} \times (0.0 + 0.118) \times 0.5 = 2.6 \times 10^{-4}$ and zero for a damage threshold at either the 100% and 105% RST levels. Since the probability of damage (i.e., 0.118) is a point estimate evaluated at discrete PGV levels, the probability over the frequency range is assigned the average value. The probability of damage evaluations, assuming a threshold at the 90% RST level, are conservatively used in the final summary to provide additional conservatism.

Similarly, the estimated probability of damage to a codisposal waste package from impacts is given in Table 4.1-6, assuming a damage threshold at the 90% RST level, for PGV values between 0.4 and 4.07 m/s inclusively and, assuming a damage threshold at the 100% RST level, for PGV values between 2.44 and 4.07 m/s inclusively. Combining the information from Table 6.4-6 and Table 4.1-6 results in a probability of damage to a codisposal waste package of $(0.29 \times (0.0 + 0.03) + 0.56 \times (0.03 + 0.56) + 0.091 \times (0.56 + 0.94) + 0.0044 \times (0.94 + 1.0)) \times 0.5 = 0.24$ and $(0.091 \times (0.0 + 0.147) + 0.0044 \times (0.147 + 0.412) \times 0.5 = 7.9 \times 10^{-3}$, assuming a damage threshold at the 90% and 100% RST levels, respectively. The estimated probability of damage from impacts for a codisposal waste package is zero, assuming a damage threshold at the 105% RST level.

From Section 6.3.2, the probability of a potentially critical configuration resulting from an assembly misload (loading curve violation) of a PWR TAD canister was evaluated as $0.014 \times 1.18 \times 10^{-5} = 1.65 \times 10^{-7}$ per canister. Likewise, the probability of absorber plate misloads was evaluated in Section 6.3.2 as 1.25×10^{-7} per canister.

The probability of the initiating event for the suite of evaluations for the vibratory impact event in the seismic scenario is given above as 2.6×10^{-4} for the commercial SNF TAD waste packages and 0.24 for the codisposal waste packages. Evaluating the event sequences for commercial SNF and DOE-owned SNF with potential for criticality using the number of waste packages given in Table 4.1-3 as 4,568 for 21-PWR TAD canisters, as 2,915 for 44-BWR TAD canisters, and as 1,223 for DOE-owned SNF canisters with criticality potential (DOE1, DOE2, and DOE7 groups) for seismic vibratory induced impact damage, assuming a damage threshold at the 90% RST level, and noting that the seismic probability results from a Binominal evaluation, gives:

PWR TAD canister loading curve violation:
$2.6 \times 10^{-4} \times \{1-P_B\,(0; (1.65 \times 10^{-7}), 4568)\} = 2.0 \times 10^{-7}$

PWR TAD canister absorber misload:
$2.6 \times 10^{-4} \times \{1-P_B\,(0; (1.25 \times 10^{-7}), 4568)\} = 1.5 \times 10^{-7}$

44-BWR TAD canister absorber misload:
$2.6 \times 10^{-4} \times \{1-P_B\,(0; (1.25 \times 10^{-7}), 2915)\} = 9.5 \times 10^{-8}$

DOE-owned SNF canister absorber misload (DOE1, DOE2, and DOE7):
$0.24\, \{1-P_B\,(0; (1.25 \times 10^{-7}), 1223)\} = 3.7 \times 10^{-5}$.

Evaluating the event sequences for DOE-owned SNF with the additional absorber loading constraint from Section 4.1.15 that the DOE1 (MOX) and DOE7 waste form (aluminum-based DOE-owned SNF) include neutron absorber shot as well as plate type absorbers that eliminates these waste forms from the set that has potential for criticality results in an estimated DOE-owned SNF canister absorber misload probability given by:

DOE-owned SNF canister absorber misload (89 DOE2 canisters, Table 4.1-2):
$0.24 \times \{1-P_B\,(0; (1.25 \times 10^{-7}), 89)\} = 2.7 \times 10^{-6}$.

Thus, a conservative estimate for the probability of achieving a configuration with criticality potential in the repository resulting from seismic vibratory induced impact damage, assuming a damage threshold at the 90% RST level, with subsequent SCC breaching of the waste package OCB for commercial SNF and DOE-owned SNF, based on summing this set of events, including the DOE1, DOE2, and DOE7 contributions is 3.7×10^{-5} for 10,000 years. The estimate, including only the DOE2 contribution, is 3.1×10^{-6} for 10,000 years. These results have been developed on a very conservative basis (e.g., use of damage probabilities at the 90% RST level and a maximum of five intervals to represent the seismic hazard curve). As stated in Section 6.3.2, the probabilities evaluated from the complete event sequences are expected to be significantly lower than from using a truncated sequence of events to estimate the probability of achieving a configuration with potential for criticality. For example, using a maximum of 35 intervals in the hazard curve for estimating the probability of impact damage to codisposal waste packages reduced the estimated probability of vibratory impact damage to the codisposal waste packages by approximately 20% (output DTN: MO0705CRITPROB.000, file: *CSNF TAD & CDSP WP Impact damage.xls*).

Evaluating the probability with Equation 6.3-1 that at least one of the seismic vibratory events occur that induce impact damage to the commercial SNF and DOE-owned SNF waste package OCB, assuming a damage threshold at the 100% RST level, gives:

PWR TAD canister loading curve violation:
0.0

PWR TAD canister absorber misload:
0.0

44-BWR TAD canister absorber misload:
0.0

DOE-owned SNF canister absorber misload (DOE1, DOE2, and DOE7):
$7.9 \times 10^{-3} \times \{1-P_B\ (0; (1.25 \times 10^{-7}), 1223)\} = 1.2 \times 10^{-6}$.

Evaluating the event sequences for DOE-owned SNF with the additional absorber loading constraint from Section 4.1.15 that the DOE1 (MOX) and DOE7 waste form (aluminum-based DOE-owned SNF) include neutron absorber shot as well as plate type absorbers that eliminates these waste forms from the set that has potential for criticality results in an estimated DOE-owned SNF canister absorber misload probability given by:

DOE-owned SNF canister absorber misload (89 DOE2 canisters, Table 4.1-2)
$7.9 \times 10^{-3} \times \{1-P_B\ (0; (1.25 \times 10^{-7}, 89)\} = 8.8 \times 10^{-8}$.

Thus, a conservative estimate for the probability of achieving a configuration with criticality potential in the repository resulting from impact damage, assuming a damage threshold at the 100% RST level, from a seismic vibratory event with subsequent SCC breaching of the waste package OCB for commercial SNF and DOE-owned SNF, including the DOE1, DOE2, and DOE7 contributions is 1.2×10^{-6} for 10,000 years. The estimate, including only the DOE2 contribution, is 8.8×10^{-8} for 10,000 years.

The critical configuration is potentially achievable for waste packages for which either an absorber misload or loading curve violation has occurred in combination with either: 1) a schoepite-moderated system as a result of water vapor entry through SCCs or 2) a flooded configuration resulting from drip shield breach and subsequent OCB failure due to localized corrosion. These probability evaluations have been developed for the in-package degraded scenario, FEP 2.1.14.19.0A (Table 1.2-1). The events in the in-package intact configuration scenario, FEP 2.1.14.18.0A, are the same as those for the in-package degraded scenario and do not increase the probability of achieving a configuration with potential for criticality. The probability values for FEP 2.1.14.18.0A are thus insignificant.

A seismic vibratory impact induced breach of a waste package is not expected to increase the criticality potential for the near-field or for the far-field configurations (FEPs 2.1.14.20.0A and 2.2.14.10.0A, respectively) since the waste package breach from a seismic vibratory impact is limited to SCCs that do not permit sufficient accumulation for criticality in the external environment. A discussion of the events required for external critical configurations is provided in Section 4.1.15 with the conclusion that the likelihood for the occurrence of configurations with potential for criticality was very low. Thus, the criticality potential in the near-field and far-field locations referenced by FEPs 2.1.14.20.0A and 2.2.14.10.0A from a seismic vibratory impact induced breach of a waste package is insignificant.

6.4.2.2 Evaluation of Drip Shield Tearing and Rupture from Seismic Vibratory Ground Motion Events

Drift Degradation Analysis (BSC 2004 [DIRS 166107], Section 6.1.4) states that the structural response of the rock in the repository to thermal and mechanical loadings is governed by the fracture geometry and properties. The quantity of lithophysae in the rock correlates inversely with the number of fractures having a trace length greater than 1 meter. This intrinsic difference between nonlithophysal and lithophysal rock units results in different failure modes. The fracture surfaces in the nonlithophysal rock units provide the primary weaknesses in the system and control the resulting block dimensions (BSC 2004 [DIRS 166107], Section 6.4.1.1). In contrast, the intense small-scale fracturing in the lithophysal rock combined with the presence of lithophysae distributed approximately uniformly through the rock results in a relatively weaker material, such that, when stressed beyond its limits, is expected to crumble into relatively small block sizes controlled by the spacing of natural fractures (BSC 2004 [DIRS 166107], Section 6.4.1.1).

A rockfall event that could potentially result in drip shield damage is a function of the size of the rock block, the impact velocity and drip shield impact location. The drip shields may accumulate damage from rockfall induced by vibratory ground motion from repository closure until the drip shield plates fail. In the lithophysal units, the accumulation of rubble from multiple seismic events and the dynamic motion during a seismic event may generate damaged areas on the drip shield. These damaged areas are regions that exceed the residual (tensile) stress threshold for the drip shield plates, potentially leading to a network of stress corrosion cracks that could allow seepage water to migrate through the cracks. In the nonlithophysal units, rock blocks can impact the drip shield in an unfilled or partly filled drift. Block impacts may result in damaged areas on the drip shield plates and, in more extreme cases, may result in tearing or rupture of the plates and failure of the axial stiffeners beneath the crown of the drip shield (SNL 2007 [DIRS 176828], Section 6.10).

Drip shield damage is expected to be primarily limited to stress corrosion cracking (SNL 2007 [DIRS 176828], Section 6.1.4). Drip shield separation during a seismic event is improbable since significant rockfall is very likely as complete collapse of the emplacement drifts in lithophysal rock is expected for exceedance frequencies less than 1.0×10^{-5} per year at a PGV of approximately 2 m/s (BSC 2004 [DIRS 166107], p. 6-170; SNL 2007 [DIRS 176828], Section 6.7.3.1). Relatively small amounts of rockfall tend to prevent drip shield separation, as demonstrated in *Mechanical Assessment of the Drip Shield Subject to Vibratory Motion and Dynamic and Static Rock Loading* (BSC 2004 [DIRS 169753], Section 5.3.3.1). However, an analysis of drip shield damage due to the impact of large rock blocks from the nonlithophysal geologic units indicates that drip shield damage may be sufficiently extensive such that the rock block could contact the waste package (SNL 2007 [DIRS 178851], Section 6.4.7.1). This event is evaluated in Section 6.4.3.3.

The probabilities of the waste package OCB failing during the 10,000-year period following repository closure from drip shield rupture and localized corrosion (Section 4.1.14), are evaluated in Output DTN: MO0705CRITPROB.000, File: *DSLC 01-29-08.zip,* Folders 3D and 4D for nonlithophysal and lithophysal units, respectively. These files, derived from DTN: MO0712PANLNNWP.000 ([DIRS 184480], file: *mo0712panlnnwp 000.zip*, folders: 3D and 4D), were extended to include the probabilities peculiar to criticality for this analysis and shown in Output DTN: MO0705CRITPROB.000], file: *DSLC 01-29-08.zip,* folders: 3D and 4D.

From Section 6.3.2, the probability of a potentially critical configuration resulting from an assembly misload (loading curve violation) of a PWR TAD canister was evaluated as $0.014 \times 1.18 \times 10^{-5} = 1.65 \times 10^{-7}$ per canister. Likewise, the probability of absorber plate misloads was evaluated in Section 6.3.2 as 1.25×10^{-7} per canister. The probabilities of occurrence of configurations with potential for criticality due to loading curve violations or absorber misloads over the 10,000-year period are listed in Table 6.4-7.

Table 6.4-7 Probability of Criticality due to Seismic Vibratory Events Resulting in Drip Shield Rupture and Waste Package Failure from Localized Corrosion

Criticality Event Sequence	Probability of Waste Package OCB Failure – Lithophysal Zone	Probability of Waste Package OCB Failure – Nonlithophysal Zone	Total Probability
PWR TAD Loading Curve Violation	5.60×10^{-10}	2.66×10^{-11}	5.9×10^{-10}
PWR TAD Canister Absorber Misload	4.24×10^{-10}	2.02×10^{-11}	4.4×10^{-10}
BWR TAD Canister Absorber Misload	2.71×10^{-10}	1.29×10^{-11}	2.8×10^{-10}
DOE-owned SNF Canister Absorber Misload [a]	2.70×10^{-10}	5.87×10^{-12}	2.8×10^{-10}
DOE-Owned SNF Canister Absorber Misload [b]	1.97×10^{-11}	3.90×10^{-13}	2.0×10^{-11}

Output DTN: MO0705CRITPROB.000], file: *DSLC 01-29-08.zip*, folders 3D and 4D.

[a] Includes DOE-owned SNF waste form groups DOE1, DOE2, AND DOE7.
[b] Includes only DOE-owned SNF waste form group DOE2.

PWR = Pressurized Water Reactor, BWR = Boiling Water Reactor, OCB = outer corrosion barrier.

Thus, a conservative estimate for the probability of achieving a configuration with criticality potential in the repository resulting from a seismic vibratory induced drip shield rupture and subsequent localized crevice corrosion breaching of the waste package OCB for commercial SNF and DOE-owned SNF, including the DOE1, DOE2, and DOE7 contributions is 1.6×10^{-9} for 10,000 years. The estimate, including only the DOE2 contribution, is 1.3×10^{-9} for 10,000 years. These probability evaluations have been developed for the in-package degraded scenario, FEP 2.1.14.19.0A (Table 1.2-1). The events in the in-package intact configuration scenario, FEP 2.1.14.18.0A, are the same as those for the in-package degraded scenario and do not increase the probability of achieving a configuration with potential for criticality. The probability values for FEP 2.1.14.18.0A are thus insignificant.

A seismic vibratory rockfall and localized corrosion induced breach of a waste package would permit degradation and transport of fissile material into the external environment. However, such an event is not expected to increase the criticality potential for either the near-field or far-field configurations (FEPs 2.1.14.20.0A and 2.2.14.10.0A, respectively) since the probability of a waste package breach from a seismic vibratory rupture of a drip shield is already very low and external accumulation can only proceed after such an event. A discussion of the events required for external critical configurations is provided in Section 4.1.15 with the conclusion that the likelihood for the occurrence of configurations with potential for criticality was very low. Thus, the criticality potential in the near-field and far-field locations referenced by FEPs 2.1.14.20.0A and 2.2.14.10.0A from a seismic vibratory drip shield rupture and localized corrosion induced waste package breach is insignificant.

6.4.2.3 Evaluation of Waste Package Damage from Seismically Induced Large, Single Rock Block Falls

This section evaluates the probability of achieving a configuration with criticality potential in the repository resulting from large, single block impacts to the waste package (after penetration of the drip shield resulting from structural failure) in the nonlithophysal rock units. Impacts that can damage a waste package must fail the drip shield stiffeners that have different fragility characteristics than the drip shield plates (DTN: MO0703PASDSTAT.001 [DIRS 183148], file: *Frame Fragility Analysis.xls*). The large block analysis indicated that waste package damage could occur for the most severe events involving rock Block 1 (SNL 2007 [DIRS 178851], Section 6.4.7.3) characterized by a rock block mass of 28.29 metric tons at a PGV level of 5.35 m/s (SNL 2007 [DIRS 178851], Table 6-153). It is important to note that the analysis for Block 1 was based solely on the 5.35 m/s PGV level that corresponds to an exceedance frequency of 1×10^{-7} per year on the unbounded hazard curve (Table 4.1-5) but is well below the 1×10^{-8} annual exceedance frequency on the bounded hazard curve that is the basis for TSPA. The conclusion from the calculations at a PGV level of 5.35 m/s is that rock block 1 would cause the stiffeners to fail. The maximum stiffener displacement expected for drip shield stiffeners from an impact of rock Block 1 (28.29 metric tons) is 20.4 cm for a drip shield (SNL 2007 [DIRS 178851], Section 6.4.7.3). Since the impact is expected to fail the drip shield stiffeners, there is a possibility that deformation of the drip shield may continue such that contact between the rock block and waste package OCB could happen although at a substantially reduced velocity but still sufficient to be an initiator for SCCs in the waste package OCB. However, a complete failure process of the drip shield has yet to be performed for the impact of rock block 1 (which fails the drip shield stiffeners). (Note: This failure mode is screened out for TSPA (SNL 2007 [DIRS 179476], FEP 1.2.03.02.0B) on the basis that failure of the drip shield stiffeners is unrealistic.)

Failure of the drip shield plates from impacts by rock blocks 2 through 7 do not cause contact between the drip shields and the waste packages because the axial stiffeners do not tear or rupture (SNL 2007 [DIRS 176828], Table 6-51). Thus, there is no potential for damage to the waste packages from rupture of the drip shield plates due to impacts by rock blocks 2 through 7 because the framework of the drip shields remains structurally intact (i.e., the axial stiffeners remain intact) and are able to deflect rockfall debris away from the waste packages (SNL 2007 [DIRS 176828], Section 6.10.2.11).

Although a rockfall event that could fracture the drip shield stiffeners is hypothetically possible, the probability of such an event is well below the low probability limit for the bounded hazard curve (Figure 4.1-1). Thus, the contribution of such events to the probability of achieving a configuration with criticality potential is considered insignificant in all the repository locations for FEPs 2.1.14.18.0A, 2.1.14.19.0A, 2.1.14.20.0A, and 2.2.14.10.0A (Table 1.2-1).

6.4.3 Consequences of Seismic Faulting Events for Waste Packages

Results from analyses of waste package damage due to fault displacement during a seismic event are documented in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6.11.7). As noted in Section 6.11.1.3 of *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828]), the clearances in Table 6.4-1 are based upon the drip shields remaining intact over the first 10,000 years after repository closure since failure of the drip shield is not expected to occur until sometime during the long time scale required for peak dose assessment. Thus, the clearances in Table 6.4-1 are appropriate for criticality evaluations, which are also limited to the first 10,000 years after repository closure (Section 4.2.2.1) when the drip shields are expected to remain intact. It is stated in *Seismic Consequence Abstraction* (SNL 2007 [DIRS 176828], Section 6.11) that since the dose related to fault displacement is expected to be a small fraction of total dose, detailed calculations of the structural response of EBS components to fault displacements are not warranted for TSPA-LA (SNL 2007 [DIRS 176828], Section 6.11). For criticality analyses, the same methodology as used for TSP-LA is applied, but represented with a finer level of detail than used for the damage abstraction for fault displacement responses in the TSPA-LA (SNL 2007 [DIRS 176828], Section 6.11.7). The calculations for the criticality analysis are given in Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," derived from DTN: MO0705FAULTABS.000 [DIRS 183150] updated to the waste package inventory from Table 4.1-4.

Fractional lengths of the various waste package types in the inventory, which are used to determine the expected number of waste package failures from faulting, are listed in Table 6.4-9. Table 4.1-4 provides the expected number of waste packages by type that are emplaced on each fault. Tables 4.1-5 and 4.1-6 show the result of combining the exceedance frequencies that cause failure and the number of packages emplaced on faults in Table 4.1-4 to determine the cumulative number of waste packages expected to fail by type as a function of annual exceedance frequency.

Table 6.4-8. Fractional Length per Waste Package Variant

Waste Package Type	Nominal Quantity	Total Length of Waste Package Type (mm)	Fraction of Waste Packages (% of Total Length)
Commercial SNF TAD Canister	7,483	4.378×10^{7}	74.7
Codisposal Short	1,600	5.916×10^{6}	10.1
Codisposal Long	1,474	7.818×10^{6}	13.3
Codisposal MCO	210	1.109×10^{6}	1.9

Sources: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," rows 132 to 140.

MCO = multicanister overpack.

The mean annual seismic exceedance frequency of concern with respect to the probability of criticality evaluation ranges from 10^{-4} to 10^{-8} per year as given in Section 4.1.1. The frequency for occurrence of a disruptive seismic event is estimated from a Poisson frequency basis for TSPA to be in the range of 4.3×10^{-4} per year that results in a typical interval between events of approximately 2,300 years (SNL 2007 [DIRS 176828], Section 6.1.3). The Poisson frequency is the difference in two values (i.e., $\lambda_{min} - \lambda_{max}$) that is given by ($4.3 \times 10^{-4} - 1 \times 10^{-8}$) per year) = 4.3×10^{-4} per year (SNL 2007 [DIRS 176828], Section 6.12.2). However, it is noted that a frequency of this magnitude may be appropriate for TSPA use but is high for criticality evaluations since only the more-severe (low-frequency) seismic events with faulting can cause sufficient damage to waste packages to affect their criticality potential.

Seismic consequences for criticality evaluations have been evaluated for events with annual exceedance frequencies ranging from approximately 10^{-7} to 10^{-8} per year (SNL 2007 [DIRS 176828], Table 6-65). Note that the severity of seismic events is inversely proportional to the exceedance frequency. For seismic events with an annual exceedance frequency greater than 1.2×10^{-7} per year (i.e., less-severe earthquakes), no waste package damage is expected to occur due to faulting as shown in Tables 4.1-5 and 4.1-6. For seismic events with an annual exceedance frequency less than approximately 1.2×10^{-7} per year (i.e., more-severe earthquakes) waste package failure from seismically induced faulting is initiated. The number of failed waste packages increases with increasing seismic energy (decreasing annual exceedance frequency) to a maximum number that depends on waste package variant as shown in Tables 4.1-5 and 4.1-6. The overall PGV range with respect to seismic faulting events for the commercial SNF TAD and codisposal waste packages is subdivided into three or four subranges for this analysis, depending on the waste package variant as shown in the column labeled "PGV Value" in Table 6.4-9 for each waste package variant. The probabilities of these basic events are determined with Equation 6.3-1 and the information provided in Table 6.4-9:

Table 6.4-9. Probabilities of Seismic Faulting Events with Waste Package Failure Capability

Commercial SNF TAD Waste Package Variant					
PGV Value (m/s)	**λ_1 (events/year)**	**λ_2 (events/year)**	**t_1 (years)**	**t_2 (years)**	**Probability**
4.07 to 3.77	1.0×10^{-8}	2.7×10^{-8}	10,000	0	1.7×10^{-4}
3.77 to 3.41	2.7×10^{-8}	7.0×10^{-8}	10,000	0	4.3×10^{-4}
3.41 to 3.34	7.0×10^{-8}	8.2×10^{-8}	10,000	0	1.2×10^{-4}
Codisposal Waste Package Variant					
4.07 to 4.00	1.0×10^{-8}	1.3×10^{-8}	10,000	0	3.0×10^{-5}
4.00 to 3.62	1.3×10^{-8}	4.1×10^{-8}	10,000	0	2.8×10^{-4}
3.62 to 3.21	4.1×10^{-8}	1.1×10^{-7}	10,000	0	6.9×10^{-4}
3.21 to 3.18	1.1×10^{-7}	1.2×10^{-7}	10,000	0	1.0×10^{-4}

Source: Output DTN: MO0705CRITPROB.000, file: *Fault Displacement Abstraction for Criticality Updated DTN 10-25-07.xls*, spreadsheet: "Tables by WP Type," rows 197 to 211.

The sequence of events begins with the occurrence of a seismic faulting event with sufficient displacement to breach a waste package. Events requiring probability values for the calculation are listed as follows:

1. Probability of a seismic faulting event over an exceedance range where sufficient displacement can shear waste packages

2. Number of failed waste packages for a seismic faulting event

3. Probability of improper absorber material in a TAD or DOE-owned SNF canister

4. Probability of a loading curve violation for a PWR TAD canister.

The mean probability of a seismic faulting event is a point value derived from the probability of a seismic event with faulting as given in Table 6.4-9 multiplied by the incremental number of waste packages with criticality potential being impacted within each frequency range given in Tables 4.1-5 and 4.1-6. The probabilities of the remaining events in this scenario, with the exception of the presence of water, are derived from preclosure activities, making those values independent of the postclosure time period. The probability of installing improper absorber plate material in a TAD canister is a fabrication related error given in Table 4.1-1 as 1.25×10^{-7} per canister. An analysis of commercial SNF misload probabilities was documented in *Commercial Spent Nuclear Fuels Waste Package Misload Analysis* (BSC 2003 [DIRS 166316]). Results from this analysis assign the probability of misloading an SNF assembly into a 21-PWR Absorber Plate Waste Package as 1.18×10^{-5} (BSC 2003 [DIRS 166316], Table 41). However, neighboring assemblies that have low reactivity values may provide partial compensation for the excess reactivity from the incorrectly loaded assembly. Given that a misload occurs, the likelihood of the misloaded configuration having potential for criticality has been shown to be 0.014 from results of a probabilistic calculation of that potential (SNL 2008 [DIRS 182788], Section 7). The cited analysis is used as a surrogate for misloading a TAD canister since the misloading of an assembly into a TAD canister requires the same improper selection of an assembly with characteristics (burnup and enrichment) in the unacceptable range of the loading curve. Thus, the total misload probability important for criticality for a PWR TAD canister is $0.014 \times 1.18\times10^{-5} = 1.65 \times 10^{-7}$ per TAD canister.

The probability of the initiating event is a function of the PGV range as given in Table 6.4-9. The fraction of the total number of waste packages that are PWR TAD canisters is 4,568/7,483, the fraction of 44-BWR TAD canisters is 2,915/7,483, and the fraction of DOE-owned SNF (DOE1, -2, and -7) canisters is 1,223/3,074 from Table 4.1-2. Evaluating the probability of seismic faulting damage for commercial SNF and DOE-owned SNF waste packages with potential for criticality using Equation 6.3-1 (that at least one of the seismic faulting event occurs) with the fractions of waste package variant combined with the total number of failed waste packages over the entire repository within the PGV range gives:

PWR TAD canister loading curve violation:
$1.2 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.65 \times 10^{-7}, (19.4 \times 4568/7483)) + 4.3 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.65 \times 10^{-7}, (27.6 - 19.4) \times 4568/7483)) + 1.7 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.65 \text{ x } 10^{-7}, (32.1 - 27.6) \times 4568/7483)) = 6.3 \times 10^{-10}$

PWR TAD canister absorber misload
$1.2 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (19.4 \times 4568/7483)) + 4.3 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (27.6 - 19.4) \times 4568/7483)) + 1.7 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (32.1 - 27.6) \times 4568/7483)) = 4.8 \times 10^{-10}$

44-BWR TAD canister absorber misload
$1.2 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (19.4 \text{ x } 2915/7483)) + 4.3 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (27.6 - 19.4) \times 2915/7483)) + 1.7 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (32.1 - 27.6) \times 2915/7483)) = 2.9 \times 10^{-10}$

DOE-owned SNF canister absorber misload (DOE1, DOE2, and DOE7)
$1.0 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (2.6+3.5) \times 1223/3074)) + 6.9 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (3.7\text{-}2.6 + 4.9\text{-}3.5) \times 1223/3074)) + 2.8 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (4.3\text{-}3.7 + 5.7\text{-}4.9) \times 1223/3074)) + 3.0 \text{ x } 10^{-5} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (21.6\text{-}4.3 + 28.5\text{-}5.7) \times 1223/3074)) = 8.1 \times 10^{-11}$.

Evaluating the event sequences for DOE-owned SNF with the additional absorber loading constraint from Section 4.1.15 that the DOE1 (MOX) and DOE7 waste form (aluminum-based DOE-owned SNF) include neutron absorber shot as well as plate type absorbers results in an estimated DOE-owned SNF canister absorber misload probability given by:

DOE-owned SNF canister absorber misload (89 DOE2 canisters, Table 4.1-2)
$1.0 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (2.6+3.5) \times 89/3074)) + 6.9 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (3.7\text{-}2.6 + 4.9\text{-}3.5) \times 89/3074)) + 2.8 \times 10^{-4} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (4.3\text{-}3.7 + 5.7\text{-}4.9) \times 89/3074)) + 3.0 \times 10^{-5} \times (1\text{-}P_B\ (0;\ 1.25 \times 10^{-7}, (21.6\text{-}4.3 + 28.5\text{-}5.7) \times 89/3074)) = 3.8 \times 10^{-12}$.

Thus, a conservative estimate for the probability of achieving a configuration with criticality potential in the repository resulting from a seismic faulting initiating event for commercial SNF and DOE-owned SNF, including the DOE1, DOE2, and DOE7 contributions is 1.5×10^{-9} for 10,000 years. The estimate, including only the DOE2 contribution, is 1.4×10^{-9} for 10,000 years.

The probability of criticality as a result of a misload in the above calculations inherently assumes that the system has adequate moderation to support criticality.

As discussed in Section 6.3.2, additional events having probabilities less than one, some much less than one, and individually or in combination likely reduce the probability that a seismic faulting event can lead to conditions needed to support criticality.

Although the waste package configuration is susceptible to waste form degradation and accumulation in the external environment, a seismic faulting event initiating a breach of a waste package is not expected to increase the criticality potential for the near-field or for the far-field configurations (FEPs 2.1.14.20.0A and 2.2.14.10.0A, respectively). The accumulation of fissile material in the external environment depends on a number of events, the first of which is the occurrence of the seismic event that has a low probability, followed by degradation, separation of fissile material from neutron absorber material, transport of the material from the waste package, and accumulation in the external location. As stated above, the probabilities of these additional events are less than one. Section 6.2.2 discusses the minimum fissile mass necessary for criticality external to the waste packages and concludes that, for a subset of the waste forms, insufficient fissile material can collect over the first 10,000-year postclosure period to achieve a critical mass. A discussion of the events required for external critical configurations is provided in Section 4.1.15 with the conclusion that the likelihood for the occurrence of configurations with potential for criticality was very low. Thus, the criticality potential in the near-field and far-field locations referenced by FEPs 2.1.14.20.0A and 2.2.14.10.0A from a seismic faulting event is concluded to be insignificant.

The events in the short sequences are considered as the principal contributors to the probability of occurrence of configurations having criticality potential following a seismic initiating event. Extending the sequences to include additional events would further decrease the probability for the occurrence of configurations with potential for criticality. Conditions inherent in the use of one or two sequences of events to estimate a conservative value for the probability of achieving a configuration with potential for criticality were discussed in Section 6.3.2. When the probabilities, although not explicitly quantified, of each of these necessary events are considered, together with the probability of the initiating event, the probability of criticality resulting from this seismic scenario is considered sufficiently low such that, if evaluated, would not change the conclusion, based on low probability, that a criticality event in the repository can be screened from further consideration in analyses.

6.5 FEPS ASSOCIATED WITH ROCKFALL EVENT SEQUENCE INITIATORS

The repository horizon lies within the Topopah Spring Tuff, and essentially consists of two main types of rock: the nonlithophysal rock and the lithophysal rock (Table 4.1-4). The nonlithophysal rocks, which comprise 15% of the emplacement area, are hard, strong, jointed rock masses whereas the lithophysal rocks, which comprise 85% of the emplacement area, are relatively more deformable with lower compressive strength (BSC 2004 [DIRS 166107], p. vii). The lithophysal rocks also contain cavities in the rock (lithophysae) that are connected by intense fracturing (BSC 2004 [DIRS 166107], Sections 6.1.2 and 6.4.1.1). Rockfall has been conjectured to be an initiating event that could cause drip shield failure through rupture leading to subsequent waste package breaching through localized corrosion (SNL 2007 [DIRS 179476], FEP 2.1.07.01.0A). Such breaches may allow the influx of seepage water (either advective or diffusive) into the waste package, which, in turn, has the potential to initiate processes leading to a critical configuration.

Three mechanisms in the repository environment have been identified as potential initiators of rockfall events in the emplacement drifts: (1) seismic vibratory ground motions, (2) thermal stress (generated by the decay heat from the emplaced waste packages), and (3) static fatigue from nominal degradation of rock (BSC 2004 [DIRS 166107], p. viii). Drip shield damage from rockfall induced by thermal loading is found to be minor since the block sizes for such rockfall are small with a mean mass of less than 0.2 metric tons (BSC 2004 [DIRS 166107], p. 6-102). The nominal case for drift degradation (i.e., considering thermal and time-dependent effects, but excluding seismic effects) results in only partial collapse of the emplacement drifts at 20,000 years. The conclusion for the nominal scenario is that negligible drift degradation will occur over the initial 10,000-year postclosure period (BSC 2004 [DIRS 166107], p. x). Thus, seismically induced rockfall is the only one of the three mechanisms that has potential for causing significant damage to the drip shields and waste packages. The probability of achieving a configuration with criticality potential in the repository resulting from a seismic vibratory induced drip shield rupture due to rockfall was evaluated in Section 6.4.3.

The rockfall disruptive event criticality FEP scenarios are identified as FEPs 2.1.14.21.0A, 2.1.14.22.0A, 2.1.14.23.0A, and 2.2.14.11.0A (Table 1.2-1). The scenarios associated with FEPs are: (1) in-package criticality resulting from a rockfall event (intact configuration), (2) in-package criticality resulting from a rockfall event (degraded configuration), (3) near-field criticality resulting from a rockfall event, and (4) far-field criticality resulting from a rockfall event. The probability of the occurrence of configurations with criticality potential for these scenarios is insignificant since no failures of the drip shield are expected from the nonseismically initiated rockfall events and, thus, no damage to waste package OCBs is expected.

6.6 FEPS ASSOCIATED WITH IGNEOUS EVENT SEQUENCE INITIATORS

The igneous disruptive event (intersection of the license application repository footprint by a volcanic dike or dike system) is described in *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada* (BSC 2004 [DIRS 169989]). The annual frequency of igneous disruptive events is characterized in the cited reference by a probability distribution having a mean value of 1.7×10^{-8} per year as given in Section 4.1.4. The mean frequency value of 1.7×10^{-8} per year translates to a probability of 1.7×10^{-4} for the regulatory period [10,000 years following closure of the repository (40 CFR Part 197, Subpart B [DIRS 184076], Section 36)].

6.6.1 Waste Package Failure from an Igneous Disruptive Event

The beginning of the volcanic activity is typically characterized by effusive (liquid) magma flow or pyroclastic flow (clots of melt in a stream of gas, where the overall concentration of volatile species in basalts around the repository is estimated to range from 1 wt % to 4 wt %) into the drifts and/or aerial expulsions (SNL 2007 [DIRS 177430], Section 1.3.2). The igneous disruption scenario can be subdivided for purposes of criticality evaluations into two segments. The first is an intrusive scenario where an igneous basaltic dike (magma-filled crack) intersects one or more repository drifts. Any magmatic intrusion into the repository region has the potential for continued movement to the surface. The second igneous scenario (designated as extrusive) is an event where the magma dike extends through the repository to the ground surface and transports radioactive waste onto the surface and (possibly) into the atmosphere (SNL 2007 [DIRS 177431]). The extrusive scenario assumes that volcanic eruptions in the Yucca Mountain region are of the violent Strombolian type for the entire duration of the explosive phase (SNL 2007 [DIRS 177431], Section 5.1.1). This scenario represents the most violent type of Strombolian activity, in which the near-vent ballistic component is minimal and tephra dispersal in a wind-blown convective plume dominates, maximizing the dispersal of contaminants for such activity. Given that an igneous disruptive event occurs in the repository, there is an estimated 0.28 probability (SNL 2007 [DIRS 177432], Table 7-1) of at least one extrusive center forming within the repository boundary. This translates to a mean frequency of ($1.7 \times 10^{-8} \times 0.28 =$) 4.8×10^{-9} per year for development of at least one extrusive center (i.e., conduit) in the repository. Given the violent nature of an extrusive igneous event, it is expected that the fuel material would be dispersed, and hence the probability of a criticality from such an event is insignificant.

The igneous scenario at Yucca Mountain is based on the observation that most basaltic eruptions begin as fissure eruptions, discharging magma where a dike intersects the earth's surface (SNL 2007 [DIRS 177430], Figure 1-1). The marginal distribution for the number of extrusive conduits that may be formed during the disruptive igneous scenario ranges from zero through 13, with one being the most probable number as given in Section 4.1.4. Basaltic magma is transported from a region of melting in the earth's mantle to the earth's surface through dikes. In the Yucca Mountain region, dikes are typically 1- to 12-m wide (Section 4.1.4). *Dike/Drift Interactions* (SNL 2007 [DIRS 177430], Section 1.3.2) states that during magma ascent and decompression, volatile gases such as water vapor and carbon dioxide exsolve and increase the volume of the magma. The resulting expansion drives the basaltic magma farther through the upper few kilometers of crust. Because volatiles may be concentrated near the crack tip of the ascending magma, the start of volcanism is typically characterized by pyroclastic eruptions (volcanic explosions and aerial expulsion of clastic rock from a volcanic vent) of gas-rich magma. Although the behavior of an ascending dike beneath the repository could be influenced by a number of factors, the conceptual model assumes that the dike propagates through the repository (SNL 2007 [DIRS 177430], Section 1.3.2). When the magma within the dike reaches the level of the repository, magma will be available to flow into drifts. Because the entry of magma from the dike into the drift is not necessarily instantaneous with intersection, it is unlikely that dike intersection will result in an abrupt explosive entrance of magma into the drift (SNL 2007 [DIRS 177430], Section 1.3.2).

As magma reaches the repository level and emplacement drifts, it will start flowing into the drifts. Magma is expected to first enter the drift in the form of pyroclastic flow as a result of rapid exsolution and expansion of gases into the atmospheric pressure in the drift. Pyroclastic flow is usually a violent event in which gases carrying particles of magma expand to high velocities. [Note: Such a flow through a relatively small gap (drifts with significant rockfall) will certainly cause deformation and erosion of material in the gap that will be carried into un-intersected drifts. Material eroded and entrained by pyroclastic magma will reduce its temperature and slow its progress, and will most likely close the gap reducing or preventing further flow down the emplacement drifts.]

The temperatures of the waste package, the canister internals, and the SNF will increase to near-magma temperatures within days to weeks (SNL 2007 [DIRS 177430], Figure 6-94) exceeding 700°C for one to nineteen months, depending on the temperature of the magma and the radioactive decay heat generated by the waste (SNL 2007 [DIRS 177430], Section 6.4.6). At these high waste package temperatures, Fe–Zr and Ni–Zr liquid eutectics are expected to form (starting at approximately 948°C (ASM International 1996 [DIRS 181641], Fe–Zr and Ni–Zr phase diagrams). Depending on whether the waste package is initially cold or hot, the analyses indicate waste package peak temperatures ranging from approximately 1,250 K (977°C) (SNL 2007 [DIRS 177430], Figure 6-105) to over 1,400 K (1,127°C) (SNL 2007 [DIRS 177430], Figure 6-94). This will produce high pressures within the waste packages. The presence of magma in the drift will cause the fuel element temperature to increase rapidly. Initially, as the temperature increases, the fission gas pressure within the fuel will increase. This may eventually cause extensive cladding damage releasing the fission gas. This type of damage is expected to begin at temperatures well below the Fe-Zr eutectic temperature.

Noncorroded waste packages are not expected to rupture from the internal pressures; however, the heat of the magma will cause the waste packages to lose strength and external pressure is expected to cause plastic deformation, resulting in the waste package shells collapsing onto their internal components. However, if a package were corroded or otherwise weakened it would have a lower tensile strength that could be exceeded by the internal pressure, providing a possible mechanism for package breaching (SNL 2007 [DIRS 177430], Section 6.4.8.3.1).

As the temperature of the waste packages increase above the eutectic temperature, zirconium and iron will react at contact locations to form a low-melting-point eutectic liquid. That is, a molten Zr–Fe alloy will form at the points where Zircaloy components of the SNF assemblies are in contact with the Stainless Steel Type 316L SNF canister and basket structure. As stated previously, eutectic liquid formation will occur at temperatures of approximately 948°C, which is significantly below the melting point of either Zircaloy (1,850°C) or stainless steel (1,538°C). The oxide film on the fuel assembly results in only a minor delay in the formation of the eutectic as the oxide films are expected to dissolve into the bulk zircaloy on the timescale of hours at the predicted magma temperatures in the Yucca Mountain repository.

The major constituents of the liquid eutectic would be zirconium and iron, with the melt approximately 84% zirconium and 16% iron at the eutectic temperature. The composition ratio varies with temperature and, near the maximum expected magma temperature, the ratio of zirconium to iron can reach 80% to 20%. Because of the large inventory of zirconium present in the SNF, it is expected that a large amount of zirconium–iron eutectic will form over the time the waste package internals are above the eutectic temperature.

Several of the DOE-owned SNF fuel types incorporate neutron absorber material that is necessary for criticality control for certain degraded scenarios (Section 4.1.15). The neutron absorber is provided by basket material made of a nickel-gadolinium alloy and/or gadolinium-bearing shot composed of iron or aluminum. Temperatures will also be sufficiently high such that the DOE aluminum fuels and gadolinium-containing aluminum shot used with certain DOE fuels for criticality control are expected to melt. Thus, these configurations are susceptible to fuel and absorber material reconfiguration by melting. This could be a result of the basket structure slumping due to the high temperature environment from the surrounding magma or from the formation of a mass either by melting or eutectic formation. These effects leading to configurations where the fissile material is concentrated away from the bulk of the neutron absorber in the canisters will be unlikely. Melting or eutectic formation or slumping will always provide some mixing between the fissile materials and the neutron absorber.

Cladding materials for commercial spent fuel in waste packages exposed to the high temperatures and corrosive gases at high pressures in the magmatic environment for one to tens of months are not expected to remain intact but to become comminuted due to oxidation (SNL 2007 [DIRS 177430], Section 6.4.9). High-level wastes in the DHLW canisters are not expected to be substantially changed by any intrusion. While little is known about chemical reactions that might occur between basalt and waste forms, the basalts are expected to include silicate and oxide minerals, although salts can not be ruled out (SNL 2007 [DIRS 177430], Section 6.4.8.3.5). However, the chemistry of the seepage water entering and passing through the basalt filled drifts is changed by chemical reactions between basalt and the original formation water with the expected result of altering percolation water in the basalts with respect to pH values and ionic strengths (SNL 2007 [DIRS 177430], Section 6.6.7). The stability of waste forms and the dissolution of radionuclides in the water are affected by water chemistry, and results of studies of in-package chemistry show that the chemistry of in-package solutions is buffered such that the chemistry of the influent water has little effect on the pH and ionic strength of the exfluent water (SNL 2007 [DIRS 180506], Section 6.6.2[a]) (e.g., all of the waters considered in *In-Package Chemistry Abstraction* (SNL 2007 [DIRS 180506], Figures 6-14[a] and 6-20[a]) end up with nearly the same pH and ionic strength as they exit the waste package).

Once the repository cools to below 100°C, the presence of liquid water or humid air may cause the solidified eutectic mass to corrode. Due to the uncertainty in the composition of the solidified mass, its corrosion properties are not known. Given the uncertainty in the corrosion resistance of the solidified eutectic phase mixture of zirconium, uranium, iron, nickel, chromium, oxygen, and other trace elements, the solidified eutectic may corrode and completely oxidize. Since the majority of the solidified eutectic is zirconium, the corrosion product is predominantly ZrO_2, the stable oxide of zirconium, with a volume that is 50% greater than that of the metal. Furthermore, due to the large strains experienced by the brittle oxide during corrosion, the eutectic oxide may contain a significant amount of porosity.

The above conditions, coupled with the corrosive nature of the magmatic environment, are all deleterious to the integrity of the drip shields and waste packages, and breaching of the barriers is expected to be ubiquitous. Therefore, as is stated in *Dike/Drift Interactions* (SNL 2007 [DIRS 177430], Section 6.4.8.3.5), the fraction of drip shields expected to fail is one (1) and the fraction of waste packages expected to fail is also one (1) (i.e., they will no longer function as a barrier to seepage or radionuclide transport).

As the magma starts to cool, it will solidify with the progressive crystallization inward from the cooler drift surface since there are no free surfaces and the heat flow will primarily be directed radially outward. Since the edges of a drift will already be solid and there will be a plastic layer between the cooler, solid basalt and the interior, analyses indicate that as the magma starts to crystallize, the entrained gases collect in the remaining fluid phase with the pressure increasing as the fluid volume decreases. At temperatures near 1,100°C, creep strains could exceed the strain limits of the lid materials causing failure of the waste package. As a result, magma could possibly enter the waste package (SNL 2007 [DIRS 177430], Section 6.4.8.3). Such conditions are expected to affect all waste packages in magma flooded emplacement drifts because the source of pressure is inherent in the magma itself. As the magma cools further within the waste package, more gas will be released and vesicles will form. The end result of the possible intrusion of magma into waste packages would be having the free volumes filled with vesicular basalt (i.e., containing isolated millimeter-to-centimeter size spheroidal voids) with little, if any large internal voids. Given the expected composition of magmatic volatiles and a nominal 10 MPa pressure, it is concluded that waste packages can be subjected for several months to a highly corrosive vapor consisting of steam, CO_2, SO_2, H_2, HCl, H_2S, CO, S_2, and HF at temperatures of 900°C to 1,100°C (SNL 2007 [DIRS 177430], Section 6.4.9).

Temperatures in the invert fill supporting the pallets and waste packages will approach or exceed the glass transition temperature for the crushed tuff of approximately 550°C to 650°C for as long as three years after intrusion if the waste packages are hot at the time of intrusion; for cold waste packages and a hot, effusive intrusion, temperatures in the upper half of the invert fill may exceed glass transition temperature for a few months. The glass transition temperature is the value where the rock becomes plastic and would not provide support to the pallets or waste packages (SNL 2007 [DIRS 177430], Section 6.4.7.2.7). Under such conditions, changes in orientation (rotational and/or axial) of waste packages can be conjectured to occur as the support structures fail (SNL 2007 [DIRS 177430], Section 6.4.8.3.2). However, due to the uncertainty in evaluating the magmatic effects on waste package integrity (SNL 2007 [DIRS 177430], Section 6.4.9), a high degree of uncertainty exists for any particular configuration and the criticality analysis of igneous events focuses on conditions necessary for criticality instead of specific configurations (SNL 2007 [DIRS 181373], Section 6.2).

Although it is quite probable that flow of magma into drifts will result in plastic deformation of waste packages, displacement and relocation relative to their initial positions in the drift is not expected to occur. An analysis of the requisite magma velocity to accomplish any significant movement of waste packages indicated that such velocities must exceed 15 to 20 meters per second (SNL 2007 [DIRS 177430], Section 6.4.8.3.3) while magma velocities are estimated to be in the range of 1 to 5 meters per second (SNL 2007 [DIRS 177430], Section 6.4.8.3.2). Thus, it is improbable that any significant movement of waste packages from their emplacement positions will occur.

The waste forms and absorber materials are expected to remain inside the waste packages following an igneous event (SNL 2007 [DIRS 177430], Section 6.4.8.3.1) but some degraded configurations of the waste form may decrease the effectiveness of the absorber. In addition, *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007 [DIRS 181165], Section 6.2.2) indicates that sensitization of stainless steel and borated stainless steel can occur during heating and cooling, such as would occur from magmatic intrusion. In principle, heating of the stainless steel to magmatic temperatures might cause sensitization and a reduction in corrosion resistance. During sensitization, the chemical composition in the vicinity of the grain boundaries can be altered by the precipitation of chromium-containing carbides, which depletes chromium at the edges of the adjacent alloy grains (typically austenite) and increases potential for intergranular corrosion, since the chromium-depleted regions fail to produce a chromium-oxide passivating layer. Subsequent slow cooling at 500°C to 750°C may desensitize the steel, as chromium diffuses back into the depleted zones. However, the situation at still lower temperatures is less clear, as the solubility of the carbide phase decreases. Fox and McCright (1983 [DIRS 159344]) argue that heating in the repository for years, at temperatures of 350°C and below, may cause desensitization, especially in Stainless Steel Type 304 alloys.

The increase in general corrosion rates due to sensitization is not well-quantified. Most literature is concerned with intergranular corrosion and cracking. However, no credit is taken for the structural integrity of the steel parts after an igneous event, so intergranular corrosion is of limited importance. Kain et al. (1994 [DIRS 182348], Table 2) show that the corrosion rate of sensitized Stainless Steel Type 304L is no more than a factor of 3.7 times the corrosion rate of the "as-received" Stainless Steel Type 304L, under the same extreme conditions (boiling acid).

The Stainless Steel Type 304B does not suffer sensitization in the same way that Stainless Steel Type 304L is affected. The metal borides are actually boro-carbides of the form $(Cr,Fe)_2(B,C)$ or $(Cr,Fe)_{23}(B,C)_6$ and effectively soak up most excess carbon. The borides precipitate at rather high temperatures and are stable down to fairly low temperatures, so there is no formation of chromium carbide. For heat-treated Stainless Steel Type 304B, Moreno et al. (2004 [DIRS 179295]) conclude:

> …it is not possible to talk about a common sensitized state as no carbides are found at the grain boundaries.

The pitting potential for heat-treated Stainless Steel Type 304B was approximately the same as for the as-received material, which is not the case for nonborated Stainless Steel Type 304. However, in these relatively short tests, a chromium-depleted region may have formed around the boride particles, enhancing the chance that the boride grains might break free from a corroding surface, under conditions where a concomitant oxide corrosion product is slow to form (SNL 2007 [DIRS 181165], Section 6.2.2).

In summary, it is expected that an igneous intrusion would sufficiently compromise the integrity of the waste packages, drip shields, and cladding in affected emplacement drifts to make them ineffective (i.e., a total loss of function in isolating waste packages and waste forms from seepage water when it returns after drifts have cooled). Thus, it is improbable that a bathtub configuration (forming a closed-bottom container necessary for pooling) can be maintained or even created in a post-igneous intrusion environment. However, water is expected to percolate through disrupted waste packages facilitating conversion of the fissile material to a moderated form of uranium (schoepite and metaschoepite).

6.6.2 Consequences of an Intrusive Igneous Event for Commercial SNF and DOE-SNF

The igneous disruptive event criticality FEP scenarios are identified as FEPs 2.1.14.24.0A, 2.1.14.25.0A, 2.1.14.26.0A, and 2.2.14.12.0A (Table 1.2-1). The scenarios associated with these FEPs are: (1) in-package criticality resulting from an igneous event (intact configuration), (2) in-package criticality resulting from an igneous event (degraded configuration), (3) near-field criticality resulting from an igneous event, and (4) far-field criticality resulting from an igneous event. The criticality potential of the intact configuration is negligible since the discussion in Section 6.6.1 argues that intact configurations cannot be maintained in drifts that are impacted by igneous intrusions. The criticality potential of in-package degraded configuration is negligible since criticality analyses of potential configurations have shown that these configurations are less reactive than the design basis configuration (SNL 2007 [DIRS 181373], Section 7), provided that the absorber material is not misloaded and, for TAD canisters, commercial SNF assemblies have not been misloaded. An evaluation of the criticality potential has been performed for the DOE-owned SNF waste forms in codisposal waste packages affected by an igneous intrusion event in the emplacement drifts. The calculation focused on in-package configurations derived from the base configuration where the waste package was fabricated and loaded according to specifications. The results of this calculation showed that the k_{eff} values of the waste package for all configurations are less than the waste form critical limit during and after an igneous intrusion event for all DOE-owned SNF representative waste forms (BSC 2006 [DIRS 181335], Section 7-10).

Support for the conclusion that the criticality potential of in-package degraded configurations is negligible is evidence showing that the boron in borated stainless steel has a very low solubility within the iron matrix of the steel (He et al. 2000 [DIRS 181597], p. 218; Goldschmidt 1971 [DIRS 181593], p. 911; Sourmail et al. 2004 [DIRS 181595], p. 1,275). Instead of a solid solution, the boron is present as separate chromium boride particles, with a composition of $(Cr_2Fe)_{7.66}(B,C)_6$ (Moreno et al. 2004 [DIRS 179295], p. 577). These particles do not dissolve into the aqueous solution during degradation of the steel but are left behind as insoluble products during corrosion (Fix et al. 2004 [DIRS 171745], p. 126; Lister et al. 2007 [DIRS 182177], pp. 39 to 43). Likewise, the gadolinium in the DOE-owned SNF canisters forms phosphate or carbonate corrosion products that have very low corrosion rates (Section 4.1.15).

6.6.3 Summary of an Intrusive Igneous Event

Conditions inherent in the use of one initiating event to estimate a conservative value for the probability of achieving a configuration with potential for criticality were discussed in Section 6.3.2. Additional event sequences, for which the probability is less than or equal to one but not quantifiable, that are necessary for a configuration to have criticality potential in the igneous disruptive scenario beyond those listed in Section 6.3.2 are as follows:

- Immediate or delayed waste package damage; - The combined effects of plastic deformation, an enhanced corrosive environment, phase transformation, and ordering reactions (e.g., atomic dislocations or slippage), causing embrittlement and increased susceptibility to localized corrosion suggest that the waste packages will fail rather rapidly with respect to the time scale of interest (10,000 years).

- Separation of fissionable material from the neutron absorber material or lack of absorber material; - During the igneous intrusive event and immediately following, temperatures will be elevated for a sufficiently long period to induce thermal creep and reconfiguration of the internal components. However, there is no expectation that most of the components or materials in commercial SNF waste packages will relocate from their locations relative to each other. This is reasonable given that intrusion temperatures do not exceed the melting temperatures of the majority of the waste package (SNL 2007 [DIRS 177430], Section 6.4.8.3) or waste form component materials (melting temperature of UO_2 is approximately 2,600°C (Todreas and Kazimi 1990 [DIRS 107735]), p. 306), with the exception of the eutectics that are self limiting.

  An evaluation documented in *DOE SNF Material Interaction Potentials during an Intrusive Igneous Event at Yucca Mountain* (Smith and Loo 2007 [DIRS 183392], Section 11) concluded there was potential for some reconfigurations in most of the nine criticality fuel groups if immersed in magma. Results from the evaluation indicated there is no mechanism that can lead to reconfiguration of mixed oxide waste forms. Other waste forms, particularly aluminum-based SNF and uranium metal SNF, are likely to exhibit some geometry changes. However, results of criticality analyses all of the representative DOE-owned SNF waste forms in waste packages affected by an igneous intrusion showed that, for the range of reconfigurations considered, none exceeded the critical limit for the particular waste form (BSC 2006 [DIRS 181335], Section 7.10) provided the SNF canisters were fabricated and loaded according to specifications.

During an igneous disruptive event water, silica, and carbon are the only potential moderating materials for internal and external configurations. Carbon is not present within the magma composition, and the amount of silicon necessary to act as a moderator combined with its relatively low moderating effectiveness is insufficient to support criticality for low enriched systems. The physical and chemical environment around the waste package and waste form materials in contact with active magma will include abundant steam and other potentially corrosive or reactive volatiles (SNL 2007 [DIRS 177430], Section 6.4.8.3) where the estimated water content of potential magmas at Yucca Mountain ranges from 1.0 wt % to 5.0 wt % with a uniform probability (DTN: LA0612DK831811.001 [DIRS 179987], file: *LA0612DK831811_001*, spreadsheet: EPAR_TPO-13Jan07). As stated previously, temperatures could be in the range of 700 < T (°C) < 1,200 for several months. The vapor pressures can be on the order of 7 MPa giving a vapor density approximately double the density at atmospheric pressure. Thus, the water density during the igneous event will be too low to moderate the neutrons sufficiently to result in a criticality event for low enriched systems. In addition, the waste package internals are designed to preclude criticality when fully flooded with full density water. This fact coupled with the fact that as fuel temperatures increase, the resonance absorption increases, which causes low enriched systems to decrease in reactivity, indicates that a criticality event during the igneous event is very improbable (i.e., physical conditions necessary are not present to support criticality). Therefore, the probability of sufficient moderating material to support criticality during an igneous event has been considered sufficiently low such that, if quantified, would not significantly increase the overall probability of criticality in the repository.

Criticality analyses of the representative DOE-owned SNF high-enriched uranium waste forms in waste packages affected by an igneous intrusion showed that none exceeded the critical limit for the particular waste form (BSC 2006 [DIRS 181335], Section 7.10). The various configurations considered in the analysis included degraded wet and dry conditions but not dispersal of the waste form (BSC 2006 [DIRS 181335], Section 6.3).

Upon cooling of the magma, the prototypical igneous scenario with all uranium atoms that was represented, among other parameters, as fully hydrated schoepite at full saturation, was less reactive than the design basis as illustrated in *Commercial Spent Nuclear Fuel Igneous Scenario Criticality Evaluation* (SNL 2007 [DIRS 181373], Section 6.2). These conditions indicate that the probability evaluations are over-estimating the actual probabilities (calculated values are based on a system that is fully flooded with full density water) associated with the events considered, and would require consideration (i.e., probabilities) of additional events within a sequence (e.g., how much schoepite actually forms and its saturation level). These would be less than one, and sometimes much less than one, thus further decreasing the overall event sequence probability.

An igneous intrusion is not expected to increase the probability of occurrence for configurations with criticality potential for far-field scenarios (FEP 2.2.14.12.0A). Neither is an intrusion event expected to increase similar probabilities for near-field scenarios (FEP 2.1.14.26.0A) but, more likely, reduce the probability of occurrence for configurations with criticality potential for near–field scenarios since the drift will be filled with magma limiting the presence of liquid water as well as mobility of fissile material. In addition, the temperatures in the invert fill supporting the pallets and waste packages will approach or exceed the glass transition temperature for the crushed tuff. This is expected to result in formation of tuff vitrophyre up to 3 m from the contact surface (SNL 2007 [DIRS 174260], Section F.2.2), which would result in a significantly reduced void fraction within the invert available for fissile material accumulation. Section 6.2.2 discusses the factors necessary for criticality to occur external to the waste packages and concludes that insufficient fissile material can collect over the first 10,000-year postclosure period to achieve a critical mass.

The impact of an intrusive igneous event on waste packages and various SNF types has been evaluated for configurations with criticality potential (i.e., presence of fissile material, neutron moderator, lack of neutron absorbers) by considering a representative configuration in lieu of attempting to evaluate a range of specific environmental parameters and configurations, along with an estimate of their probability of occurrence, that could generate a large number of possible event sequences and outcomes. The single representative configuration is considered representative of ones having criticality potential following an initiating intrusive igneous event and provides a basis for demonstrating the additional events and processes that would be required to result in criticality following an intrusive igneous event. A detailed criticality assessment of configurations for the various DOE-owned SNF waste forms has been performed (SNL 2007 [DIRS 181373]; BSC 2006 [DIRS 181335]) with all configurations shown to be subcritical. In addition, a qualitative evaluation of the additional events and processes that would be required for criticality has been developed as discussed in Section 6.3.2.

The specific geometry and composition of the numerous intermediate configurations are dependent on the environmental conditions and cannot all be defined individually for analysis. Considering the increased variability in the potential geometric reconfigurations, effects on material performance, and neutron spectrum changes resulting in varied neutron absorber effectiveness, the numbers would be of limited value considering the high degree of uncertainty associated with any given scenario that may be evaluated. The initiating event probability for the igneous intrusive event (1.7×10^{-4}) is already a factor of 1,400 below the probability of seismic vibratory ground motion damaging the codisposal waste package (0.24) as developed in Section 6.4.2. Therefore, considering the probability values associated with the conditions necessary for criticality discussed previously (e.g., absorber misload, assembly misload) the resultant probability of criticality resulting from this disruptive igneous scenario is considered sufficiently low such that, if evaluated, would not change the conclusion, based on low probability, that a criticality event in the repository can be screened from further consideration in analyses.

7. CONCLUSIONS

7.1 SUMMARY OF PROBABILITY EVALUATIONS

Results of the event sequences evaluated for the nominal criticality FEP scenario, seismic disruptive FEP scenario, rockfall disruptive FEP scenario, and igneous disruptive FEP scenario are shown in Table 7.1-1 which summarize the values from Sections 6.3, 6.4, 6.5, and 6.6 calculated as conservative estimates for their contributions to the probability of achieving a configuration with criticality potential in the repository over the initial 10,000-year period following closure.

Table 7.1-1. Estimated Probability of Criticality Configurations in the Repository over 10,000 Years

Waste Package Variant	**In-Package Intact**	**In-Package Degraded**	**Near-Field**	**Far-Field**
	Probability Estimate for FEPs Associated with Nominal (Early Failure) Event Sequence Initiators (Section 6.3.2)			
PWR TAD canister	Insignificant	1.5×10^{-7}	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	4.1×10^{-8}	Insignificant	Insignificant
DOE-owned SNF canister [a]	Insignificant	1.7×10^{-8}	Insignificant	Insignificant
DOE-owned SNF canister [b]	Insignificant	1.3×10^{-9}	Insignificant	Insignificant
SubTotal	NA	2.1×10^{-7}	NA	NA
	Probability Estimate for FEPs Associated with Seismic Event Sequence Initiator - Vibratory Impact at 90% RST (Section 6.4.2.1)			
PWR TAD canister	Insignificant	3.4×10^{-7}	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	9.5×10^{-8}	Insignificant	Insignificant
DOE-owned SNF canister [a]	Insignificant	3.7×10^{-5}	Insignificant	Insignificant
DOE-owned SNF canister [b]	Insignificant	2.7×10^{-6}	Insignificant	Insignificant
SubTotal	NA	3.7×10^{-5}	NA	NA
	Probability Estimate for FEPs Associated with Seismic Event Sequence Initiator - Vibratory Drip Shield Rupture (Section 6.4.2.2)			
PWR TAD canister	Insignificant	1.0×10^{-9}	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	2.8×10^{-10}	Insignificant	Insignificant
DOE-owned SNF canister [a]	Insignificant	2.8×10^{-10}	Insignificant	Insignificant
DOE-owned SNF canister [b]	Insignificant	2.0×10^{-11}	Insignificant	Insignificant
SubTotal	NA	1.6×10^{-9}	NA	NA
	Probability Estimate for FEPs Associated with Seismic Event Sequence Initiator - Single Block Rockfall (Section 6.4.2.3)			
PWR TAD canister	Insignificant	Insignificant	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	Insignificant	Insignificant	Insignificant
DOE-owned SNF canister [a]	Insignificant	Insignificant	Insignificant	Insignificant
DOE-owned SNF canister [b]	insignificant	Insignificant	Insignificant	Insignificant
SubTotal	NA	NA	NA	NA

Table 7.1-1. Estimated Probability of Criticality Configurations in the Repository over 10,000 Years (Continued)

Waste Package Variant	In-Package Intact	In-Package Degraded	Near-Field	Far-Field
	Probability Estimate for FEPs Associated with Seismic Event Sequence Initiator - Faulting (Section 6.4.3)			
PWR TAD canister	Insignificant	1.1×10^{-9}	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	2.9×10^{-10}	Insignificant	Insignificant
DOE-owned SNF canister [a]	Insignificant	8.1×10^{-11}	Insignificant	Insignificant
DOE-owned SNF canister [b]	Insignificant	3.8×10^{-12}	Insignificant	Insignificant
SubTotal	NA	1.5×10^{-9}	NA	NA
	Probability Estimate for FEPs Associated with Rockfall Event Sequence Initiator (Section 6.5)			
All Waste Package Variants	Insignificant	Insignificant	Insignificant	Insignificant
	Probability Estimate for FEPs Associated with Igneous Event Sequence Initiator (Section 6.6.2)			
PWR TAD canister	Insignificant	Insignificant	Insignificant	Insignificant
44-BWR TAD canister	Insignificant	Insignificant	Insignificant	Insignificant
DOE-owned SNF canister	Insignificant	insignificant	Insignificant	Insignificant
Total [a]	Insignificant	3.7×10^{-5}	Insignificant	Insignificant
Total [b]	Insignificant	3.3×10^{-6}	Insignificant	Insignificant
Total [c]	Insignificant	6.3×10^{-7}	Insignificant	Insignificant

[a] DOE-owned SNF waste forms DOE1, DOE2, and DOE7 without distributed neutron absorber in shot form.
[b] DOE-owned SNF waste form DOE2 without distributed neutron absorber in shot form.
[c] Distributed neutron absorber in all DOE-owned SNF with criticality potential.

Source: Output DTN: MO0705CRITPROB.000, file: *"Prob Calc."*

NA = not applicable.

Using the available geologic repository and engineered barrier systems information, and surrogate evaluations based on the best available information, a conservative value for the total probability of achieving a configuration with criticality potential was estimated as 3.7×10^{-5} over the 10,000-year period following repository closure. The total estimated probability of achieving a configuration with criticality potential becomes 3.3×10^{-6} over the 10,000-year period following repository closure, provided that DOE-owned SNF canisters for MOX SNF (DOE1) and aluminum-based SNF (DOE7) contain a distributed neutron absorber (Section 4.1.15). The estimated total probability of achieving a configuration with criticality potential is further reduced to 6.3×10^{-7} over the 10,000-year period following repository closure if the DOE-owned SNF canisters for U-Zr Hydride (DOE2) also contain a distributed absorber.

The probability of achieving a configuration with criticality potential (3.7×10^{-5} over 10,000 years) has been developed on a very conservative basis with respect to criticality events and is below the regulatory probability criterion of 1×10^{-4} over 10,000 years. As discussed in Section 6.3.2, the probabilities evaluated from complete event sequences are expected to be significantly lower than the calculated value of 3.7×10^{-5} over 10,000 years. However, this estimate of the probability of criticality does not include the evaluation of naval SNF. The overall probability will remain below the regulatory criterion provided that the value for the naval SNF is less than ($1.0 \times 10^{-4} - 3.7 \times 10^{-5} =$) 6.3×10^{-5} for the repository over 10,000 years.

7.2 EVALUATION OF YUCCA MOUNTAIN REVIEW PLAN CRITERIA

The YMRP (NRC 2003 [DIRS 163274]) contains acceptance criteria intended to establish the basis for the review of the material contained in the license application. Because this report serves, in part, as the basis for the license application, the information contained herein conforms to applicable acceptance criteria. The acceptance criteria that are applicable to this calculation as presented in Table 4.2-1 are evaluated with respect to the method of addressing the criteria that is also listed in Table 4.2-1. The acceptance criteria for FEP screening analysis rely primarily on the collective screening tests of low probability, but regulations also allow for exclusion of a FEP on the basis of low consequence or if the process is specifically excluded by the regulations (Section 4.2.2).

7.3 CRITICALITY FEPS SCREENING JUSTIFICATION

The justification for screening postclosure criticality from further consideration in the repository is on low probability (Section 4.2.2.1) as the total probability bound for all configurations with criticality potential occurring in the repository is below the criterion from Section 4.2.2.1 (Table 7.1-1).

INTENTIONALLY LEFT BLANK

8. INPUTS AND REFERENCES

8.1 DOCUMENTS CITED

101978 Holman, J.P. 1997. *Heat Transfer*. 8th Edition. New York, New York: McGraw-Hill. TIC: 239954.

103896 Parrington, J.R.; Knox, H.D.; Breneman, S.L.; Baum, E.M.; and Feiner, F. 1996. *Nuclides and Isotopes, Chart of the Nuclides.* 15th Edition. San Jose, California: General Electric Company and KAPL, Inc. TIC: 233705.

107735 Todreas, N.E. and Kazimi, M.S. 1990. *Nuclear Systems I, Thermal Hydraulic Fundamentals.* New York, New York: Hemisphere Publishing. TIC: 226511.

108050 ASME (American Society of Mechanical Engineers) 1993. *Steam Tables, Thermodynamic and Transport Properties of Steam.* 6th Edition. New York, New York: American Society of Mechanical Engineers. TIC: 103243.

112178 Shoesmith, D.W. and King, F. 1998. *The Effects of Gamma Radiation on the Corrosion of Candidate Materials for the Fabrication of Nuclear Waste Packages.* AECL-11999. Pinawa, Manitoba, Canada: Atomic Energy of Canada Limited. ACC: MOL.19990311.0212.

138239 CRWMS M&O 2000. *Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams.* CAL-MGR-MD-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000214.0479.

139383 Swain, A.D. and Guttmann, H.E. 1983. *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report. NUREG/CR-1278.* Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 246563.

152658 DOE (U.S. Department of Energy) 2000. *Review of Oxidation Rates of DOE Spent Nuclear Fuel, Part 1: Metallic Fuel.* DOE/SNF/REP-054, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 248978.

152857 Chow, V.T.; Maidment, D.R.; and Mays, L.W. 1988. *Applied Hydrology*. McGraw-Hill Series in Water Resources and Environmental Engineering. New York, New York: McGraw-Hill. TIC: 217564.

153009 Reed, D.T. and Van Konynenburg, R.A. 1991. "Effect of Ionizing Radiation on the Waste Package Environment." *High Level Radioactive Waste Management, Proceedings of the Second Annual International Conference, Las Vegas, Nevada, April 28-May 3, 1991. 2*, 1396-1403. La Grange Park, Illinois: American Nuclear Society. TIC: 204272.

156140 Reed, D.T. and Van Konynenburg, R.A. 1988. "Effect of Ionizing Radiation on Moist Air Systems." *Scientific Basis for Nuclear Waste Management XI, Symposium held November 30-December 3, 1987, Boston, Massachusetts.* Apted, M.J. and Westerman, R.E., eds. *112,* 393-404. Pittsburgh, Pennsylvania: Materials Research Society. TIC: 203662.

159344 Fox, M.J. and McCright, R.D. 1983. *An Overview of Low Temperature Sensitization.* UCRL-15619. Livermore, California: Lawrence Livermore National Laboratory. ACC: HQS.19880517.2438.

163274 NRC (U.S. Nuclear Regulatory Commission) 2003. *Yucca Mountain Review Plan, Final Report.* NUREG-1804, Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards. TIC: 254568.

165482 Radulescu, H.; Moscalu, D.; and Saglam, M. 2004. *DOE SNF Phase I and II Summary Report.* DR-DSD-MD-000001 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040303.0005.

165505 YMP (Yucca Mountain Site Characterization Project) 2003. *Disposal Criticality Analysis Methodology Topical Report.* YMP/TR-004Q, Rev. 02. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: DOC.20031110.0005.

166027 DOE 2003. *Review of Oxidation Rates of DOE Spent Nuclear Fuel Part 2. Nonmetallic Fuel.* DOE/SNF/REP-068, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: DOC.20030905.0009.

166107 BSC (Bechtel SAIC Company) 2004. *Drift Degradation Analysis.* ANL-EBS-MD-000027 REV 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040915.0010; DOC.20050419.0001; DOC.20051130.0002; DOC.20060731.0005.

166316 BSC 2003. *Commercial Spent Nuclear Fuel Waste Package Misload Analysis.* CAL-WHS-MD-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20031002.0005; DOC.20050728.0004.

168133 Herrera, M.L. 2004. *Evaluation of the Potential Impact of Seismic Induced Deformation on the Stress Corrosion Cracking of the YMP Waste Packages.* SIR-04-015, Rev. 1. San Jose, California: Structural Integrity Associates. ACC: MOL.20040311.0149.

168434 DOE 2004. *Interim Report on the Corrosion Performance of a Neutron Absorbing Ni-Cr-Mo-Gd Alloy.* DOE/SNF/REP-086, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: DOC.20040412.0001.

168553 BSC 2004. *Criticality Model.* CAL-DS0-NU-000003 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040913.0008; DOC.20050728.0007.

168935 BSC 2004. *Intact and Degraded Mode Criticality Calculations for the Codisposal of TMI-2 Spent Nuclear Fuel in a Waste Package*. CAL-DSD-NU-000004 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC20040329.0002; DOC.20050601.0003; DOC.20050801.0001.

169753 BSC 2004. *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. ACC: DOC.20041028.0004; DOC.20050830.0003; DOC.20051121.0010.

169982 BSC 2004. *Aqueous Corrosion Rates for Waste Package Materials.* ANL-DSD-MD-000001 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041012.0003; DOC.20060403.0001.

169987 BSC 2004. *CSNF Waste Form Degradation: Summary Abstraction.* ANL-EBS-MD-000015 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20040908.0001; DOC.20050620.0004.

169989 BSC 2004. *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada.* ANL-MGR-GS-000001 REV 02. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041015.0002; DOC.20050718.0007.

170071 DOE 2004. *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in a Repository.* DOE/SNF/REP-090, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: MOL.20040708.0386.

170633 Kuo, K.K-Y. 1986. *Principles of Combustion.* New York, New York: John Wiley & Sons. TIC: 256280.

171745 Fix, D.V.; Estill, J.C.; Wong, L.L.; and Rebak, R.B. 2004. "General and Localized Corrosion of Austenitic and Borated Stainless Steels in Simulated Concentrated Ground Waters." *Transportation, Storage, and Disposal of Radioactive Materials, The 2004 ASME/JSME Pressure Vessels and Piping Conference, San Diego, California, USA, July 25-29, 2004.* Smith, A.C. and Hafner, R.S.; eds. PVP-Vol. 483. Pages 121-130. New York, New York: American Society of Mechanical Engineers. TIC: 256542.

171926 BSC 2004. *Intact and Degraded Mode Criticality Calculations for the Codisposal of ATR Spent Nuclear Fuel in a Waste Package.* CAL-DSD-NU-000007 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041018.0001; DOC.20050728.0002.

172494 BSC 2004. *Configuration Generator Model.* CAL-DS0-NU-000002 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20041122.0004; DOC.20050801.0004.

174260 SNL (Sandia National Laboratories) 2007. *Characterize Eruptive Processes at Yucca Mountain, Nevada*. ANL-MGR-GS-000002 REV 03. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070301.0001.

175539 BSC 2005. *Q-List.* 000-30R-MGR0-00500-000-003. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20050929.0008.

176265 NFPA 68. 2004. *Guide for Venting of Deflagrations, with Errata No. 68-02-1.* 2002 Edition. Quincy, Massachusetts: National Fire Protection Association. TIC: 258044.

176828 SNL 2007. *Seismic Consequence Abstraction.* MDL-WIS-PA-000003 REV 03. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070928.0011.

177430 SNL 2007. *Dike/Drift Interactions.* MDL-MGR-GS-000005 REV 02. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20071009.0015.

177431 SNL 2007. *Atmospheric Dispersal and Deposition of Tephra from a Potential Volcanic Eruption at Yucca Mountain, Nevada.* MDL-MGR-GS-000002 REV 03. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20071010.0003.

177432 SNL 2007. *Number of Waste Packages Hit by Igneous Events.* ANL-MGR-GS-000003 REV 03. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20071002.0001.

178236 SNL 2007. *Criticality Input to Canister Based System Performance Specification for Disposal.* TDR-DS0-NU-000002 REV 01. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070103.0002.

178519 SNL 2007. *General Corrosion and Localized Corrosion of Waste Package Outer Barrier.* ANL-EBS-MD-000003 REV 03. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070730.0003; DOC.20070807.0007.

178765 SNL 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure.* ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002; DOC.20071003.0015.

178851 SNL 2007. *Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion.* MDL-WIS-AC-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070917.0006.

178869 SNL 2007. *Technical Work Plan for: Postclosure Criticality.* TWP-EBS-MD-000018 REV 01. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070206.0003.

179295 Moreno, D.A.; Molina, B.; Ranninger, C.; Montero, F.; and Izquierdo, J. 2004. "Microstructural Characterization and Pitting Corrosion Behavior of UNS S30466 Borated Stainless Steel." *Corrosion, 60,* (6), 573-583. Houston, Texas: NACE International. TIC: 258529.

179354 SNL 2007. *Total System Performance Assessment Data Input Package for Requirements Analysis for Engineered Barrier System In-Drift Configuration*. TDR-TDIP-ES-000010 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070921.0008.

179394 SNL 2007. *Total System Performance Assessment Data Input Package for Requirements Analysis for Transportation Aging and Disposal Canister and Related Waste Package Physical Attributes Basis for Performance Assessment*. TDR-TDIP-ES-000006 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070918.0005.

179407 Wheatley, P.D. 2006. "Representation of DOE SNF and HLW in the License Application." Letter from P.D. Wheatley (INL) to R.M. Kacich (BSC) and M.K. Knowles (SNL), October 30, 2006, 0205070323, HHL:kjb. ACC: CCU.20070205.0007.

179466 SNL 2007. *Total System Performance Assessment Data Input Package for Requirements Analysis for Subsurface Facilities.* TDR-TDIP-PA-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070921.0007.

179476 SNL 2007. *Features, Events, and Processes for the Total System Performance Assessment.* ANL-WIS-MD-000026 REV 00. Las Vegas, Nevada: Sandia National Laboratories.

179567 SNL 2007. *Total System Performance Assessment Data Input Package for Requirements Analysis for DOE SNF/HLW and Naval SNF Waste Package Physical Attributes Basis for Performance Assessment*. TDR-TDIP-ES-000009 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070921.0009.

179962 SNL 2008. *Postclosure Analysis of the Range of Design Thermal Loadings.* ANL-NBS-HS-000057 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080121.0002.

180190 BSC 2007. *Waste Package Fabrication.* 000-3SS-DSC0-00100-000-001. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070315.0001.

180506 SNL 2007. *In-Package Chemistry Abstraction.* ANL-EBS-MD-000037 REV 04 ADD 01. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070816.0004.

180664 BSC 2006. *Evaluation of Neutron Absorber Materials Used for Criticality Control in Waste Packages.* CAL-DS0-NU-000007 REV 0A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20061009.0003.

181165 SNL 2007. *Geochemistry Model Validation Report: Material Degradation and Release Model.* ANL-EBS-GS-000001 REV 02. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070928.0010.

181335 BSC 2006. *Criticality Potential of Waste Packages Affected by Igneous Intrusion.* CAL-DS0-NU-000005 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20061201.0001.

181373 SNL 2007. *Commercial Spent Nuclear Fuel Igneous Scenario Criticality Evaluation.* ANL-EBS-NU-000009 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070711.0003.

181395 SNL 2007. *Geochemistry Model Validation Report: External Accumulation Model.* ANL-EBS-GS-000002 REV 01 AD 01. Las Vegas, Nevada: SNL. ACC: DOC.20071106.0015.

181403 DOE 2007. *Transportation, Aging and Disposal Canister System Performance Specification.* WMO-TADCS-000001, Rev. 0. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20070614.0007.

181533 Wheatley, P.D. 2007. "Canister Counts for Criticality Analyses for DOE SNF in the License Application." Letter from P.D. Wheatley (INL) to R.M. Kacich (BSC) and M.K. Knowles (SNL), CCN 210126, June 20, 2007. ACC: LLR.20070627.0004.

181534 BSC 2006. *21-PWR Waste Package Internal Pressure Estimate.* 000-00C-DSU0-03500-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20061108.0004; ENG.20070402.0003.

181593 Goldschmidt, H.J. 1971. "Effect of Boron Additions to Austenitic Stainless Steels Part II Solubility of Boron in 18%Cr, 15%Ni Austenitic Steel." *Journal of the Iron and Steel Institute,* Pages 910-911. London, England: Iron and Steel Institute. TIC: 259235.

181595 Sourmail, T.; Okuda, T.; and Taylor, J.E. 2004. "Formation of Chromium Borides in Quenched Modified 310 Austenitic Stainless Steel." *Scripta Materialia, 50,* 1271-1276. New York, New York: Elsevier. TIC: 259239.

181597 He, J.Y.; Soliman, S.E.; Baratta, A.J.; and Balliett, T.A. 2000. "Fracture Mechanism of Borated Stainless Steel." *Nuclear Technology, 130,* (2), 218-225. La Grange Park, Illinois: American Nuclear Society. TIC: 259236.

181641 ASM International 1996. *Binary Alloy Phase Diagrams.* 2nd Edition. Plus Updates. Version 1.0. Materials Park, Ohio: ASM International. TIC: 259552.

181678 Green, J.R. 1994. *Progress Report for the Enhancement of Radcalc: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values.* WHC-SD-TP-RPT-014, Rev. 0. Richland, Washington: Westinghouse Hanford Company. ACC: MOL.20060731.0265.

181953 SNL 2007. *Stress Corrosion Cracking of Waste Package Outer Barrier and Drip Shield Materials.* ANL-EBS-MD-000005 REV 04. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070913.0001.

182051 DOE 2007. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Rev. 19. Washington, D. C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20070717.0006.

182138 Coward, H.F. and Jones, G.W. 1952. *Limits of Flammability of Gases and Vapors.* Bulletin 503. Washington, D.C.: U.S. Government Printing Office. TIC: 241049.

182177 Lister, T.; Mizia, R.; Erickson, A.; and Trowbridge, T. 2007. *Electrochemical Corrosion Testing of Neutron Absorber Materials.* INL/EXT-06-11772, Rev. 1. Idaho Falls, Idaho: Idaho National Laboratory. ACC: LLR.20070731.0149.

182348 Kain, V.; Shinde, S.S.; and Gadiyar, H.S. 1994. "Mechanism of Improved Corrosion Resistance of Type 304L Stainless Steel, Nitric Acid Grade, in Nitric Acid Environments." *Journal of Materials Engineering and Performance, 3,* (6), 699-705. Materials Park, Ohio: ASM International. TIC: 259661.

182605 NRC (U.S. Nuclear Regulatory Commission) 1984. "Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation." Information Notice No. 84-72. Washington, D.C.: Nuclear Regulatory Commission, Office of Inspection and Enforcement. Accessed August 22, 2007. ACC: LLR.20080110.0081. URL: http://www.nrc.gov/reading-rm/doc-collection/gen-comm/info-notices/1984/in84072.html.

182643 Orrell, S.A. 2007. "Contract No. DE-AC04-94AL-85000 – Updated Lead Laboratory Recommendation for the Neutron Absorber to be Used in the Performanced-Based Requirements Document for Transport, Aging, and Disposal (TAD) Canisters." Letter from S.A. Orrell (SNL) to J.R. Dyer (DOE/OCRWM), July 9, 2007, 0712071368, with enclosure. ACC: CCU.20070712.0006.

182708 Flaherty, James E.; Fujita, Akira; Deltete, C. Paul; and Quinn, Geoffrey J. 1986. *A Calculational Technique to Predict Combustible Gas Generation in Sealed Radioactive Waste Containers.* GEND-041. Idaho Falls, Idaho: EG&G Idaho. ACC: LLR.20070919.0005.

182752 Golan, P.M. 2004. "Issuance of Audit Report No. 04-DOE-AU-004 for the Department of Energy Idaho Operations Office Spent Nuclear Fuel Program." Memorandum from P.M. Golan (DOE) to E.D. Sellers (DOE/ID), September 23, 2004, with attachment. ACC: MOL.20041214.0129; MOL.20041214.0130.

182788 SNL 2008. *CSNF Loading Curve Sensitivity Analysis.* ANL-EBS-NU-000010 REV 00. Las Vegas, Nevada: Sandia National Laboratories.

183392 Smith, R.E. and Loo, H.H. 2007. *DOE SNF Material Interaction Potentials during an Intrusive Igneous Event at Yucca Mountain.* DOE/SNF/REP-108, Rev. 0. Idaho Falls, Idaho: U.S. Department of Energy, Idaho Operations Office. ACC: CCU.20070906.0009.

183478 SNL 2008. *Total System Performance Assessment Model/Analysis for the License Application.* MDL-WIS-PA-000005 REV 00 AD 01. Las Vegas, Nevada: Sandia National Laboratories.

183927 SNL 2007. *Criticality Analyses for FFTF Fuel with Advanced Neutron Absorber Material.* ANL-DSH-NU-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories.

184078 SNL 2007. *Waste Package Flooding Probability Due to Seismic Fault Displacement.* CAL-DN0-NU-000002 REV 00C. Las Vegas, Nevada: Sandia National Laboratories.

184433 SNL 2008. *Multiscale Thermohydrologic Model*. ANL-EBS-MD-000049 REV 03 AD 02. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080201.0003.

184719 Triay, I.R. 2007. "Issuance of Audit Report No. 07-DOE-AU-002 for the Department of Energy Idaho Operations Office, National Spent Nuclear Fuel Program." Memorandum from I.R.Triay (DOE) to E.D. Sellers (DOE/ID), May 7, 2007, with attachment. ACC: MOL.20070702.0077; MOL.20070702.0078.

8.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

141257 ASME (American Society of Mechanical Engineers) 1995. "Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components." Section III of *1995 ASME Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. TIC: 245287.

168403 ASTM B 932-04. 2004. *Standard Specification for Low-Carbon Nickel-Chromium-Molybdenum-Gadolinium Alloy Plate, Sheet, and Strip.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 255846.

173375 10 CFR 71. 2005. Energy: Packaging and Transportation of Radioactive Material. ACC: MOL.20050523.0022.

176225 ANSI/ANS-8.17-2004. 2004. *American National Standard,Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.* La Grange Park, Illinois: American Nuclear Society. TIC: 257593.

176331 Regulatory Guide 3.71, Rev. 1. 2005. *Nuclear Criticality Safety Standards for Fuels and Material Facilities.* Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20060206.0325.

178058 ASTM A 887-89 (Reapproved 2004). 2004. *Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 258746.

178394 70 FR 53313. Implementation of a Dose Standard After 10,000 Years. Internet Accessible.

180319 10 CFR 63. 2007. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. Internet Accessible.

184076 40 CFR 197. 2006. Protection of Environment: Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada. Internet Accessible.

LS-PRO-0203. *Q-List and Classification of Structures, Systems, Components and Barriers.*

SCI-PRO-001. *Qualification of Unqualified Data*.

SCI-PRO-005. *Scientific Analyses and Calculations*.

SCI-PRO-004. *Managing Technical Product Inputs*.

IM-PRO-002. *Control of the Electronic Management of Information*.

IM-PRO-003. *Software Management*.

8.3 SOURCE DATA, LISTED BY DATA TRACKING NUMBER

164713 LA0307BY831811.001. Characterize Igneous Framework Additional Output. Submittal date: 07/29/2003.

172059 MO0409SPAACRWP.000. Aqueous Corrosion Rates for Non-Waste Form Waste Package Materials. Submittal date: 09/16/2004.

172682 MO0501BPVELEMP.001. Bounded Horizontal Peak Ground Velocity Hazard at the Repository Waste Emplacement Level. Submittal date: 01/11/2005.

173280 LB0407AMRU0120.001. Supporting Calculations and Analysis for Seepage Abstraction and Summary of Abstraction Results. Submittal date: 07/29/2004.

179645 MO0609SPAINOUT.002. PHREEQC Modeling Inputs and Outputs for Geochemistry Model Validation Report: External Accumulation Model. Submittal date: 09/27/2006.

179925 MO0702PASTREAM.001. Waste Stream Composition and Thermal Decay Histories for LA. Submittal date: 02/15/2007.

179987 LA0612DK831811.001. Magma and Eruption Properties for Potential Volcano at Yucca Mountain. Submittal date: 03/23/2007.

180508 MO0701PASHIELD.000. Waste Package/Drip Shield Early Failure Probabilities. Submittal date: 04/24/2007.

180946 MO0705EARLYEND.000. Waste Package/Drip Shield Early Failure End State Probabilities. Submittal date: 05/18/2007.

181380 MO0706ECTBSSAR.000. Report INL/EXT-07-12633: Electrochemical Corrosion Testing of Borated Stainless Steel Alloys. Submittal date: 06/12/2007.

181613 MO0706SPAFEPLA.001. FY 2007 LA FEP List and Screening. Submittal date: 06/20/2007.

182029 MO0703PAGENCOR.001. Output from General Corrosion and Localized Corrosion of Waste Package Outer Barrier 2007 Second Version. Submittal date: 07/18/2007.

182944 MO0604SPANOMIN.000. Nominal Case Diffusive Releases. Submittal date: 04/26/2006.

182994 MO0709TSPALOCO.000. TSPA Localized Corrosion Analysis. Submittal date: 09/13/2007.

183148 MO0703PASDSTAT.001. Statistical Analyses for Seismic Damage Abstractions. Submittal date: 09/21/2007.

183150 MO0705FAULTABS.000. Assessment of Waste Package Failure Due to Fault Displacement for Criticality. Submittal date: 09/21/2007.

183622 MO0705PHREEMOD.000. PHREEQC Input and Output Files for Geochemistry Model Validation Report: External Accumulation Model in Support of Criticality. Submittal date: 10/23/2007.

183634 MO0705SCALEGEO.000. Scale Input and Output Files for Geochemistry Model Validation Report: External Accumulation Model in Support of Criticality Calculations. Submittal date: 05/30/2007.

184480 MO0712PBANLNWP.000. Probabilistic Analysis of Non-Navy Waste Packages. Submittal date: 12/17/2007.

184848 SN0705WFLOWSCC.001. Analysis for Water Flow through Stress Corrosion Cracking (SCC) Cracks in Waste Package and Drip Shield. Submittal date: 01/25/2008.

8.4 PRODUCT OUTPUT, LISTED BY DATA TRACKING NUMBER

MO0705CRITPROB.000. Probability of Criticality. Submittal date: 02/05/2008.

8.5 SOFTWARE CODES

160873 SAPHIRE V. 7.18. 2002. WINDOWS 2000/NT 4.0. STN: 10325-7.18-00.

INTENTIONALLY LEFT BLANK

APPENDIX I
HYDROGEN DEFLAGRATION EVENTS IN A WASTE PACKAGE

APPENDIX I. HYDROGEN DEFLAGRATION EVENTS IN A WASTE PACKAGE

The consequences of hydrogen generation by radiolysis and a subsequent deflagration event in a waste package are discussed in this Appendix for potential use as a contribution to defense-in-depth for criticality purposes without an explicit probabilistic evaluation. This event is dependent on residual water being left in a TAD canister and/or waste package due to a failure of the drying and inerting process that is considered as sufficiently improbable to permit the probability to be designated as insignificant (Section 6.2).

Chemical species produced by radiolysis have been identified in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2000 [DIRS 165505], p. 2-2) as a mechanism for exacerbating corrosion of the EBS components in the repository. Radiolytic sources of corrosion have also been considered in the screening of processes affecting cladding degradation (FEP 2.1.02.15.0A) in *Features, Events, and Processes for the Total System Performance Assessment* (SNL 2007 [DIRS 179476]) and excluded on the basis of low consequences for TSPA analyses. The direct effects of radiolysis within a waste package are considered in FEP 2.1.13.01.0A (Radiolysis) and secondary effects in FEP 2.1.12.01.0A (Gas Generation) and FEP 2.1.12.08.0A (Gas Explosions in EBS) (SNL 2007 [DIRS 179476]).

The effects of radiation on fluids in either a liquid or gaseous state have been discussed by a number of authors (e.g., Green 1994 [DIRS 181678]; Shoesmith and King 1998 [DIRS 112178], p. 2). While Shoesmith and King (1998 [DIRS 112178]) address primarily the effects of gamma radiation on corrosion properties of waste package materials, such processes derive from the radiolytic effects on fluids. Such radiolytic effects on fluids may lead to formation of a variety of species such as carbon dioxide, carbon monoxide, hydrogen, oxygen, methane, and various nitrogen oxide forms. Oxidizing radicals and molecular products may be generated where oxidants include, but are not necessarily limited to, OH^+, O_2^-, H_2O_2, and O_2. Likewise, radiolysis can lead to the formation of reductants such as H^+ and H_2. In moist air environments, radiolytic processes can lead to the fixation of nitrogen as NO, NO_2, and especially HNO_3 (Reed and Van Konynenburg 1988 [DIRS 156140], pp. 393 to 404). Nitric acid is one of the principal corrosive radiolytic chemical species produced in an irradiated air-water vapor system when the hydroxyl radicals generated from the water vapor react with nitrogen dioxides, which are formed by the radiolytic reaction between nitrogen and oxygen, to form acids. The number of oxidants and reductants formed by radiolysis within intact waste packages that may contain residual moisture is limited to those that can be generated from water and/or water vapor. Since this latter group includes both hydrogen and oxygen, an analysis of the quantity and type of compounds formed, particularly potentially flammable gas mixtures such as H_2 and O_2, is necessary to evaluate an appropriate safety envelope for SNF waste packages.

Radiolytic production of particular chemical species depends upon the radiation environment, the chemical components present, and the physical environment where the radiolytic reactions are occurring. However, the yield of any given chemical species is characterized by a single parameter, "G," identified as the G-factor (Reed and Van Konynenburg 1991 [DIRS 153009], pp. 1,396 to 1,403). The "G" value represents the number of molecules of a chemical species produced per 100 eV of absorbed radiation energy in the volume containing the irradiated environment. While both gamma and neutron radiation from SNF are similarly effective with respect to radiolytic species production, the gamma dose from commercial SNF is two or more orders of magnitude larger than the neutron dose (Section 4.1.11) and thus the neutron dose contribution can be neglected.

The generation of hydrogen, which is a necessary component for flammability, from radiolytic reactions has received considerable study (e.g., Reed and Van Konynenburg 1988 [DIRS 156140]; Green 1994 [DIRS 181678]). As part of the research on radiolytic species generation, a calculational technique for generation of hydrogen in sealed waste containers was developed for a DOE program resulting in a software program, RADCALC (WHC-SA-2795-FP 1995, DOE Information Bridge Identifier, WHC-SA-2777; CONF-9506150-5). This software has been developed for calculating hydrogen concentrations in closed containers that are sealed but may exhibit a specified leak rate. The database for this program contains, in part, a compilation of net $G(H_2)$ values with references for many substances. These values are compiled from literature sources in *Progress Report for the Enhancement of RADCALC: Isotope Database, Gamma Absorption Fractions, and $G(H_2)$ Values* (Green 1994 [DIRS 181678], Table 4-2), where separate $G(H_2)$ are listed for alpha, beta, and gamma doses. The net $G(H_2)$ is based upon the total concentration of H_2 generated in an absorbing medium. The average net $G_\gamma(H_2)$ value for gamma absorption in water vapor is given as 0.49 (0.051 μmol/J) and the net $G_\gamma(H_2)$ value for liquid water is given 0.45 (0.047 μmol/J) molecules per 100 eV absorbed dose, respectively.

The net H_2 generation in a moist He volume simulating a TAD canister waste package with residual water was estimated using the $G_\gamma(H_2)$ of 0.49. The gamma source (Section 4.1.11) for the radiolysis estimate was derived from the source strength of commercial SNF assemblies in a 21-PWR waste package with 4 wt% initial ^{235}U enrichment and a 48 GWD/MTU burnup history given in Section 3.2 of the reference cited in *Total System Performance Assessment Data Input Package for Requirements Analysis for Transportation Aging and Disposal Canister and Related Waste Package Physical Attributes Basis for Performance Assessment Requirements Analysis for Subsurface Facilities* (SNL 2007 [DIRS 179394], Table 4-1, Item 03-10). Hydrogen generation from a 21-PWR TAD waste package is expected to bound similar generation from either the 44-BWR TAD canister waste package or the DOE-owned SNF canister since, on average, the radiation source strength from either of these waste package types is less (determined by thermal output) than the 21-PWR TAD canister waste package (DTN: MO0702PASTREAM.001 [DIRS 179925], spreadsheet: "Unit Cell"). Thus, it is conservative to apply the 21-PWR TAD hydrogen generation rate to the 44-BWR TAD and DOE-owned SNF canister waste packages. Considerations and caveats applicable to the calculation were as follows:

1. Sufficient water is available to permit the vapor to follow the saturated density profile.

2. Absorbed dose in a moist air medium provides a conservative estimate of absorbed dose in a moist helium environment. The conservatism results from the air-vapor density being larger than for a helium-vapor environment.

3. The G factor is independent of the gas mixture composition. The net $G\gamma(H_2)$ factor accounts for the effects of energy absorbed by helium molecules.

4. The absorbed dose varies by total density and radiolysis effects contributing to H2 generation are limited to the water vapor component.

5. Requirements for the drying of a TAD canister after closure specify that less than 0.43 moles of water remain in the TAD canister (SNL 2007 [DIRS 179394], Table 4-1, Item 04-04). Since the process steps for the purging and inerting of a waste package have not been finalized, it is expected, and thus becomes a design constraint, that verification for meeting the cited maximum water content requirement will rely on the correct performance of the drying and inerting procedure. The rationale for this constraint is that there is no identified method to determine the amount of water in a sealed waste package. Thus, undetected operational errors could occur that result in a process failure resulting in an unknown quantity of water in the waste package TAD containment structure. Thus, a failure of the purging and drying process is considered to leave sufficient water in the containment system for a deflagration event to occur (since the actual water mass is unknown), and the probability of the event can be estimated from the error probabilities for a sequence of operational processes. As stated previously, the operational process has not yet been defined limiting the analysis to generic operational event sequences. The mean probability for this type of undetected operational error based on generic event sequences is given in Table 4.1-1 as 3.84×10^{-5} per process.

Since the containment vessel is sealed, the He density will remain constant, neglecting any net volume change from thermal expansion. However, the pressure will rise with temperature. For absorbed dose, the density is the important parameter, not pressure. Thus, the helium density is held at a constant value derived from assuming the initial pressure to be two atmospheres. The helium inerting pressure in *Transportation, Aging and Disposal Canister System Performance Specification* (DOE 2007 [DIRS 181403], Section 3.1.6) is specified to be one atmosphere or greater. The total density is the sum of the partial densities of helium and water vapor. As stated in the above Bullet 4, the absorbed dose affects only the water vapor for generation of H_2, which is consistent with the net $G_\gamma(H_2)$ factor. Two H_2 generation values were calculated using the net $G_\gamma(H_2)$ factors listed above from *Progress Report for the Enhancement of RADCALC: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values* (Green 1994 [DIRS 181678], Table 4-2), which gave the same result, which supports the fourth item in the list of considerations.

The gamma dose rate in rads/hour over a time period is specified in Section 4.1.11. This rate was integrated over 10,000 years to estimate the cumulative hydrogen generation. The molecular hydrogen-to-helium volume fraction for these conditions was evaluated as being 6 vol % to 7 vol % in Output DTN: MO0705CRITPROB.000, file: *He Plus Water Vapor Density.xls*, spreadsheet: "H generation (rad)." The minimum hydrogen concentration that can support flammability varies, depending on the major constituents of the gaseous environment. A minimum hydrogen concentration of approximately 4 vol % in an air-hydrogen atmosphere at nominal (0.1 MPa) pressure is necessary to propagate a flame front (Coward et al. 1952 [DIRS 182138], Figure 7). For a helium-hydrogen environment assuming sufficient oxygen is present, the minimum hydrogen concentration that can support flammability is approximately 8 vol % (Coward et al. 1952 [DIRS 182138], Table 3). These values show that sufficient radiolytic generation of hydrogen in nonbreached TAD canisters could reach a concentration level where a pressure pulse could be generated, if ignited. Thus, radiolytic generated species cannot be á priori screened from consideration only on the basis of low concentration. However, other environmental conditions affect the ability of a flame front to develop (e.g., oxygen concentration). The minimum oxygen concentration capable of supporting a flame front is approximately 4 vol % (Coward et al. 1952 [DIRS 182138], Table 44). It is expected that minimal oxygen consumption will occur through aqueous corrosion processes within an intact waste package because of the low degradation rates of the containment materials. Thus, oxygen concentrations are expected to be proportional to the radiolytic hydrogen concentration. The effect of elevated pressure on the flammability of hydrogen-air mixtures is to increase the minimum volume percent to a limit value. However, the limit does not drop below the nominal value for atmospheric pressure so it is conservative to use the nominal value as a minimum limit (Coward et al. 1952 [DIRS 182138], Figure 5).

It is expected that development of conditions within a TAD conducive to flammability will be improbable but, if such conditions develop, the result would be likely either a propagating flame front (deflagration) or detonation since minor movements or jolting could conceivably produce an ignition source. A deflagration event is the likely mode since deflagration can be supported at a lower hydrogen concentration than a detonation. Experimental hydrogen concentration limits by volume for deflagration and detonation are, respectively, 4.6% and 15% (Kuo 1986 [DIRS 170633], Table 4.5). Based on the calculation in Output DTN: MO0705CRITPROB.000, file: *He Plus Water Vapor Density.xls*, spreadsheet: “H generation (rad),” hydrogen concentration levels are not expected to reach 15% over 10,000 years. Secondly, since the deflagration ignition source is expected to be a seismic event, if the concentration reaches the deflagration range, the event is likely to occur, effectively limiting concentration levels to the deflagration region. The important consequence with respect to criticality of a flammable event in a TAD is the resultant pressurization of the TAD. Such pressurization, whether “relatively” slow (deflagration) or rapid (detonation) (NFPA 68 2004 [DIRS 176265], Subsections 3.3.3 and 3.3.5) would result in one of two states (i.e., no breach of the waste package integrity or loss of the barrier function for the waste package). Whether the drip shield remains in place or not is of secondary importance since breaching of the waste package will allow schoepite to form.

Complete combustion of a 4.1% hydrogen mixture is estimated to heat the products to a temperature of approximately 623 K (350°C) (Coward et al. 1952 [DIRS 182138], p. 15). The design pressure rating for the waste package outer corrosion barrier is approximately 0.97 MPa (140 psia) at 650 K (707°F) (SNL 2007 [DIRS 179394], Section 4.1.2.6). The peak pressure due to 100% of fuel rods failure in a 21-PWR waste package surrogate for a TAD canister is estimated as 0.93 MPa (135 psia) at 623 K (662°F) (BSC 2006 [DIRS 181534], Table 1).

Another possible concern could be the early failure of a waste package emplaced in the subsurface repository with ineffective vacuum drying due over pressurization by steam. The waste package outer corrosion barrier is designed to meet the 1995 ASME Boiler and Pressure Vessel Code (ASME 1995 [DIRS 141257]), including internal pressures of 140 psia (0.97 MPa) (at 707°F (BSC 2007 [DIRS 180190], Appendix B, Section B4.2.2). The estimated peak pressure due to 100% fuel cladding failure at 662°F (350°C) is 135 psia (0.93 MPa); and at 752°F (400°C) is 140 psia (BSC 2006 [DIRS 181534], Section 7, Table 1).

To put these pressure levels in context, consider the incremental pressure increase from 12 liters or 12 kilograms of H_2O left in a TAD as the temperature of the waste package increases from about 75°C to 175°C, as might happen when going from normal preclosure operations to an increased temperature state shortly after repository closure.

An estimate of the pressure increase due to raising the temperature in a waste package containing 12 kg of water from 75°C to 175°C can be obtained from steam properties. At 75°C, the water will be in a two-phase state with a vapor pressure of 38,540 Pa determined by the thermodynamic properties since the volume required for a single phase vapor mass of 12 kg is approximately 50 m^3. The gas volume will remain saturated until the liquid mass is zero (i.e., dryout) that would occur at a temperature of 135°C where the vapor volume for 12 kg is 7 m^3. The vapor pressure at this point is approximately 637,550 Pa or 45.4 psi. Raising the temperature further to 175°C can be estimated using the perfect gas laws where $(p \times v/T)_1 = (p \times v/T)_2$ where $v_1 = v_2$ since the system is closed, giving a total pressure increase of approximately 50 psi. Thus, there is a possibility that a pressure increase that is on the order of the design pressure rating of the outer corrosion barrier could occur in a TAD canister.

Assuming that breach of the lid welds is the most likely failure mode, it is very improbable that either the fissile or absorber material in a properly loaded TAD will be sufficiently relocated from their nominal positions by the pressure induced failure to make a criticality event possible. The rationale for this assertion is that there is no credible mechanism, which could fragment the internal structures of an intact waste package during an internal pressurization event since the pressure source is produced by the heated gas from the flame front, which travels at subsonic speeds around the gas volume. Thus, a deflagration event does not produce a shock wave, which precludes exposure of the internal structures to large impulsive forces. Maximum to initial pressure ratios generated in a deflagration event (6% hydrogen by volume) are expected to be in the range of 1 to 4 (Coward et al. 1952 [DIRS 182138], p. 12). In addition, failure of the lids does not provide space for large internal objects (fuel rods or absorber plates) to move a significant distance since the top closure region has a steel plug in place and a complete bottom lid failure would limit any movement to a distance of less than 10 cm, the nominal waste package separation at emplacement (SNL 2007 [DIRS 179354], Table 4-4, Item 05-02). The driving force for any axial movement of waste package components is the frictional force on the objects as the gas escapes the waste package, which is unlikely to be sufficient to overcome the inertial resistance of the structures.

I.1 SCREENING ANALYSIS FOR DEFLAGRATION CASE SCENARIOS

The initiating event for this deflagration case is the occurrence of water left in a TAD canister allowing radiolytically generated hydrogen to accumulate. A deflagration event becomes possible when the concentration of hydrogen reaches the minimum deflagration concentration level. Three events requiring probability values for the bounding screening calculation are listed as follows:

1. Probability of a failure for the TAD canister drying and inerting process

2. Probability of improper absorber material in a canister

3. Probability of a loading curve violation for a PWR TAD canister.

The mean probability of a deflagration event is a point value derived from the probability of occurrence of undetected drying and inerting operational failures given in this section as 3.84×10^{-5}. The probabilities of events in this scenario are derived from preclosure activities, making those values independent of the postclosure period. The probability of installing improper absorber plate material in a TAD or DOE canister is a fabrication related error. The mean value for this type of error is given in Table 4.1-1 as 1.25×10^{-7} per canister.

An analysis of commercial SNF misload probabilities was documented in *Commercial Spent Nuclear Fuels Waste Package Misload Analysis* (BSC 2003 [DIRS 166316]). Results from this analysis assign the probability of misloading an SNF assembly into a 21-PWR Absorber Plate Waste Package as 1.18×10^{-5} (BSC 2003 [DIRS 166316], Table 41). However, neighboring assemblies that have low reactivity values may provide partial compensation for the excess reactivity from the incorrectly loaded assembly. Given that a misload occurs, the likelihood of the misloaded configuration having potential for criticality has been shown to be 0.014 from results of a probabilistic calculation of that potential (SNL 2008 [DIRS 182788], Section 7). The cited analysis is used as a surrogate for misloading waste forms in a TAD canister as the misloading of an assembly into a TAD canister requires the same improper selection of an assembly with characteristics (burnup and enrichment) in the unacceptable range of the loading curve.

The probability of misloading assemblies in the 44-BWR TAD canister is insignificant as the entire expected boiling water reactor inventory for the repository is in the acceptable region of the loading curve map (SNL 2008 [DIRS 182788], Section 6.1.1.1.3). Misloading of waste forms in DOE-owned SNF canisters is very improbable because the shape and size of the DHLW glass canisters and the various DOE-owned SNF canisters differ significantly and can be readily distinguished by visual inspection per Section 4.1.5. Thus, the waste form misload probability for DOE-owned SNF waste packages is considered insignificant.

Events such as the following have probabilities less than one, some much less than one, and individually or in combination impact the probability that the over-pressurization event can lead to configurations that have criticality potential. This list is not exhaustive nor is the supporting discussion for each complete. However, it illustrates some of the additional events in the deflagration case scenario for which information is unavailable to adequately quantify the probability, but would be necessary for achieving configurations that have criticality potential.

- Accumulation or presence of a critical mass of fissionable material – If the waste package is breached by overpressure, loss of the barrier capability of the cladding must also be assumed as proving otherwise is difficult if not impossible due to the complexity of the event. Thus, schoepite will likely form from the fissile material. The likelihood that the resultant distribution of fissile material would rearrange into a near optimum arrangement to be conducive to criticality is low as the structures are intact at the initiation of a deflagration event. Lid failure is likely to be localized to a fraction of the weld length, allowing pressure relief but preventing sizeable objects from escaping.

Thus, commercial SNF rearrangement is hypothetically possible within the assembly containment tubes, but absorber plates are required to extend the full length of the assemblies.

- Separation of fissionable material from the neutron absorber material or lack of absorber material – The waste forms and absorber materials are expected to remain inside the waste packages after a deflagration event as discussed previously. After seepage water has returned, there is little possibility of moving much of the chromium boride particles from the vicinity of the spent fuel. *Geochemistry Model Validation Report: Material Degradation and Release Model* (SNL 2007 [DIRS 181165], Section 6.3.3) indicates that, due to the boron in borated stainless steel having a very low solubility within the iron matrix of the steel, the boron is present as separate chromium boride particles instead of a solid solution. These particles do not dissolve into the aqueous solution during degradation of the steel but are left behind as insoluble products during corrosion. Therefore, the neutron absorber is expected to remain between fuel cell regions, but some degraded configurations of the waste form may decrease the effectiveness of the absorber.

- Presence of a moderator – Seepage rates and the likelihood of the waste package experiencing seepage.

I.2 SUMMARY

Gamma (and neutron) radiation from SNF can generate radiolytic hydrogen and oxygen gas in a commercial or DOE-owned SNF waste package if water is inadvertently left in the waste package or components prior to sealing the system. The gas concentrations can conceivably reach levels where a deflagration event could occur, given an ignition source. Other possible events that could affect the overall pressure in the TAD canister include steam and gas released from fuel rod cladding failure. The consequences of such an event may result in sufficient over pressurization of the vessel to cause a loss of containment integrity (i.e., a breached waste package). However, the controls for the drying and inerting processes for waste package components are expected to be similar to NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (SNL 2007 [DIRS 179394], Table 4-1, Item 04-04). Thus, it is expected that these processes will be sufficiently rigorous to reduce the likelihood of leaving residual water in the waste packages to levels that, if quantified, would not significantly alter the overall screening decision for criticality events in the repository.

INTENTIONALLY LEFT BLANK

APPENDIX II
QUALIFICATION OF EXTERNAL SOURCE DATA

APPENDIX II. QUALIFICATION OF EXTERNAL SOURCE DATA

External source data derived from Holman (1997 [DIRS 101978]), Parrington et al. (1996 [DIRS 103896]), ASME (1993 [DIRS 108050]), Coward et al. (1952 [DIRS 182138]), Kuo (1986 [DIRS 170633]), ASTM B 932-04 ([DIRS 168403], pp. 1 to 2), and ASTM A 887-89 ([DIRS 178058]) are classified as "established fact" per SCI-PRO-004, *Managing Technical Product Inputs*, as these documents are either an engineering textbook, an industry standard, or a Federal Bureau of Mines bulletin.

The radiolytic yield rate of hydrogen, $G_{\gamma}(H_2)$, used in this analysis to evaluate the effects of radiolysis in a waste package was obtained from an unqualified source. This information is "data" per SCI-PRO-004. The basis for designating this information as "data" is that the information is in a database developed for the DOE.

The inventory of DOE-owned SNF canisters and waste packages (one canister per waste package) from an unqualified source was used in this analysis. This information is classed as "data" since it superseded similar information from DTN: MO0702PASTREAM.001 ([DIRS 179925], spreadsheet: "NONCOMMERCIAL."

This section presents planning and documentation for the data qualification of the unqualified external source data used as direct input only for this analysis. Data qualifications are performed in accordance with SCI-PRO-005, *Scientific Analyses and Calculations*.

II.1 DATA FOR QUALIFICATION

There are five external sources of data used as direct input for this analysis:

1. Data for the radiolytic yield rate of hydrogen from Green 1994 [DIRS 181678] identified in Section 4.1.10

2. Data for the DOE-owned SNF canister inventory from Wheatley 2007 [DIRS 181533] identified in Table 4.1-2

3. Data on corrosion rates for neutron absorber material proposed for use with the DOE-owned SNF from DOE 2004 [DIRS 168434] identified in Table 4.1-8

4. Information on DOE-owned SNF canister and basket configurations from Smith and Loo 2007 [DIRS 183392] identified in Section 4.1.5

5. Information on requirements for neutron absorber plates in canisters for DOE-owned SNF from DOE 2004 [DIRS 170071] identified in Section 4.1.15.

II.1.1 Method of Qualification Selected

The method for qualification of the five external sources of data is the "technical assessment method." The rationale for using this method is that all three of the qualification approaches for technical assessment (SCI-PRO-001, Attachment 3, Method 5) of external source data are appropriate for consideration. Qualification process attributes used in the technical assessment of each external source are selected from the list provided in Attachment 4 of SCI-PRO-001. Attributes specifically applicable as data qualification attributes in this report are:

1. Qualifications of personnel or organizations generating the data are comparable to qualification requirements of personnel generating similar data under an approved program that supports the YMP license application process or postclosure science (#1)
2. The extent to which the data demonstrate the properties of interest (e.g., physical, chemical, geologic, mechanical) (#3)
3. Prior uses of the data and associated verification processes (#7)
4. Prior peer or other professional reviews of the data and their results (#8)
5. Extent and reliability of the documentation associated with the data (#9).

II.1.2 Technical Assessment of External Data from Green

The action taken to qualify the radiolysis data from *Progress Report for the Enhancement of Radcalc: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values* (Green 1994 [DIRS 181678]) is from SCI-PRO-001, Attachment 3, Method 5(c) as follows:

> Confirmation that the data have been used in similar applications. A discussion and documentation that the data have been used in applications similar to those for which the data will be used in this analysis. Past applications could include data used by the U.S. Nuclear Regulatory Commission or Environmental Protection Agency (or their subcontractors) in technical evaluation reports, licensing proceedings, or safety evaluation reports; by nationally/internationally recognized scientific organizations (International Atomic Energy Agency, International Radioactive Waste consortiums, etc.); or by the scientific community, including publications, peer reviews, etc.

The following criteria were used to assess the external data from *Progress Report for the Enhancement of Radcalc: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values* (Green 1994 [DIRS 181678]):

1. Qualifications of personnel or organizations generating the data are comparable to qualification requirements of personnel generating similar data under an approved program that supports the YMP license application process or postclosure science (#1)
2. Prior uses of the data and associated verification processes (#7).

Justification for the appropriate use of data from *Progress Report for the Enhancement of Radcalc: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values* (Green 1994 [DIRS 181678]):

> This document presents the results of a calculational technique for quantifying the concentration of hydrogen generated by radiolysis in sealed radioactive waste containers

developed for the DOE in a study conducted by EG&G Idaho, Inc. and the Electric Power Research Institute TMI-2 Technology Transfer Office, thus criterion 2 is satisfied. This study resulted in the report GEND-041 (Flaherty et al. 1986 [DIRS 182708]) and also resulted in a presentation to the NRC, which gained acceptance for use in ensuring compliance with NRC IE Information Notice 84-72 (NRC 1984 [DIRS 182605]) concerning the generation of hydrogen within packages, thus criterion 1 is satisfied. The data from this reference is included in Section 4.1.10.

Based on the assessment made above, data from *Progress Report for the Enhancement of Radcalc: Isotope Database, Gamma Absorption Fractions, and G(H_2) Values* (Green 1994 [DIRS 181678]) are qualified for use as direct input for this analysis.

II.1.3 Technical Assessment of External Data from Wheatley 2007

The action taken to qualify the DOE-owned SNF inventory data from Wheatley 2007 [DIRS 181533] is from SCI-PRO-001, Attachment 3, Method 5(a) as follows:

> Determination that the employed methodology is acceptable. A discussion and justification that the data collection methodology used was appropriate for the type of data under consideration (used appropriate equipment, typical of scientific and industry collection methods, etc.).

The following criteria were used to assess the external data from *Canister Counts for Criticality Analyses for DOE-owned SNF in the License Application* (Wheatley 2007 [DIRS 181533]):

1. The extent to which the data demonstrate the properties of interest (e.g., physical, chemical, geologic, mechanical) (#3)

2. Extent and reliability of the documentation associated with the data (#9)

3. Prior peer or other professional reviews of the data and their results (#8).

Justification for the appropriate use of data from: Wheatley 2007 [DIRS 181533]:

> The cited reference, Wheatley 2007 [DIRS 181533], was sent to the YMP by a manager from the National Spent Nuclear Fuel Program at the Idaho National Laboratory in support of the YMP postclosure criticality screening analysis, thus criterion 2 is satisfied. The reference contains inventory information abstracted from the DOE-owned SNF Spent Fuel Database that is maintained National Spent Nuclear Fuel Program satisfying criteria 1 and 3. Golan (2004 [DIRS 182752]) and Triay (2007 [DIRS 184719]) report that audits found the National Spent Nuclear Fuel Program was satisfactorily implementing the QARD (DOE 2007 [DIRS 182051]). The data from this reference is included in Table 4.1-2.

Based on the assessment made above, data from Wheatley 2007 [DIRS 181533] are qualified for intended use as direct input for this analysis.

II.1.4 Technical Assessment of External Data from DOE 2004

The action taken to qualify the corrosion rate data for neutron absorber material proposed for use with DOE-owned from DOE 2004 [DIRS 168434] is from SCI-PRO-001, Attachment 3, Method 5(b) as follows:

> Determination that confidence in the data acquisition or developmental results is warranted. A discussion and justification that the data acquisition and/or subsequent data development (e.g., reduction or extrapolation) discussed in source documentation was appropriate for the type of data under consideration. This could include assurances that processes were conducted by qualified professionals; data were collected under proper environmental conditions; collected results and/or data development are appropriate, reasonable, and suitable for their intended use; etc.

The following criteria were used to assess the external data from *Report on the Corrosion Performance of a Neutron Absorbing Ni-Cr-Mo-Gd Alloy* (DOE 2004 [DIRS 168434]):

1. Qualifications of personnel or organizations generating the data are comparable to qualification requirements of personnel generating similar data under an approved program that supports the YMP license application process or postclosure science (#1)

2. The extent to which the data demonstrate the properties of interest (e.g., physical, chemical, geologic, mechanical) (#3)

3. Extent and reliability of the documentation associated with the data (#9)

Justification for the appropriate use of data from DOE 2004 [DIRS 168434]:

> The cited reference, DOE 2004 [DIRS 168434], was developed for DOE under quality procedures for the National Spent Nuclear Fuel Program at the Idaho National Laboratory to help assure criticality control of DOE-owned SNF is maintained during the postclosure period of the YMP repository (Abstract), thus criteria 1 and 3 are satisfied. Golan (2004 [DIRS 182752]) and Triay (2007 [DIRS 184719]) report that audits found the National Spent Nuclear Fuel Program was satisfactorily implementing the QARD (DOE 2007 [DIRS 182051]). The reference contains information on the measured corrosion rates for the Ni-Cr-Mo-Gd Alloy developed for use as a neutron absorber for DOE-owned SNF, thus criterion 2 is satisfied. The data from this reference is included in Table 4.1-8.

Based on the assessment made above, data from DOE 2004 [DIRS 168434] are qualified for use as intended as direct input for this analysis.

II.1.5 Technical Assessment of External Data from Smith and Loo 2007

The action taken to qualify the information on DOE-owned SNF canisters and basket configurations from Smith and Loo 2007 [DIRS 183392] is from SCI-PRO-001, Attachment 3, Method 5(b) as follows:

Determination that confidence in the data acquisition or developmental results is warranted. A discussion and justification that the data acquisition and/or subsequent data development (e.g., reduction or extrapolation) discussed in source documentation was appropriate for the type of data under consideration. This could include assurances that processes were conducted by qualified professionals; data were collected under proper environmental conditions; collected results and/or data development are appropriate, reasonable, and suitable for their intended use; etc.

The following criteria were used to assess the external data from *DOE SNF Material Interaction Potentials during an Intrusive Igneous Event at Yucca Mountain* (Smith and Loo 2007 [DIRS 183392]):

1. Qualifications of personnel or organizations generating the data are comparable to qualification requirements of personnel generating similar data under an approved program that supports the YMP license application process or postclosure science (#1)

2. The extent to which the data demonstrate the properties of interest (e.g., physical, chemical, geologic, mechanical) (#3)

3. Extent and reliability of the documentation associated with the data (#9)

Justification for the appropriate use of data from Smith and Loo 2007 [DIRS 183392]:

The cited reference, Smith and Loo 2007 [DIRS 183392], was developed for DOE under quality procedures for the National Spent Nuclear Fuel Program at the Idaho National Laboratory (Section 1.2) by National Spent Nuclear Fuel Program staff, thus criteria 1 and 3 are satisfied. Golan (2004 [DIRS 182752]) and Triay (2007 [DIRS 184719]) report that audits found the National Spent Nuclear Fuel Program was satisfactorily implementing the QARD (DOE 2007 [DIRS 182051]). The reference contains information on the configurations of DOE-owned canisters and their basket geometry, thus criterion 2 is satisfied. The data from this reference is included in Section 4.1.5.

Based on the assessment made above, data from Smith and Loo 2007 [DIRS 183392] are qualified for use as intended as direct input for this analysis.

II.1.6 Technical Assessment of External Data from DOE 2004

The action taken to qualify the information on requirements for neutron absorber plates as structural elements in canisters for DOE-owned SNF from DOE (2004 [DIRS 170071]) is from SCI-PRO-001, Attachment 3, Method 5(b) as follows:

Determination that confidence in the data acquisition or developmental results is warranted. A discussion and justification that the data acquisition and/or subsequent data development (e.g., reduction or extrapolation) discussed in source documentation was appropriate for the type of data under consideration. This could include assurances that processes were conducted by qualified professionals; data were collected under proper environmental conditions; collected results and/or data development are appropriate, reasonable, and suitable for their intended use; etc.

The following criterion was used to assess the external data from *Packaging Strategies for Criticality Safety for "Other" DOE Fuels in a Repository* (DOE 2004 [DIRS 170071]):

1. Qualifications of personnel or organizations generating the data are comparable to qualification requirements of personnel generating similar data under an approved program that supports the YMP license application process or postclosure science (#1)

Justification for the appropriate use of data from DOE 2004 [DIRS 170071]:

> The cited reference, DOE 2004 [DIRS 170071], was developed for DOE under quality procedures for the National Spent Nuclear Fuel Program at the Idaho National Laboratory (Section 1.1) by National Spent Nuclear Fuel Program staff, thus criterion 1 is satisfied. Golan (2004 [DIRS 182752]) and Triay (2007 [DIRS 184719]) report that audits found the National Spent Nuclear Fuel Program was satisfactorily implementing the QARD (DOE 2007 [DIRS 182051]). The data from this reference is included in Section 4.1.15.

Based on the assessment made above, data from DOE 2004 [DIRS 170071] are qualified for use as intended as direct input for this analysis.