Generic Deep Geologic Disposal Safety Case

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Used Fuel Disposition Campaign
G. Freeze, M. Voegele,
P. Vaughn, J. Prouty,
W.M. Nutt, E. Hardin, S.D. Sevougian
Sandia National Laboratories
Argonne National Laboratory

August 2013
Revision 1
FCRD-UFD-2012-000146
SAND2013-0974P

Sandia National Laboratories

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

FCT Quality Assurance Program Document

FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	Generic Deep Geologic Disposal Safety Case / M2FT-13SN0808041
Work Package Title and Number	FT-13SN080804 GDSM
Work Package WBS Number	1.02.08.08 DR Generic Disposal System Level Modeling
Responsible Work Package Manager	Geoff Freeze (Name/Signature)
Date Submitted	

Quality Rigor Level for Deliverable/Milestone	☐ QRL-3	☒ QRL-2	☐ QRL-1 ☐ Nuclear Data	☐ N/A*

This deliverable was prepared in accordance with Sandia National Laboratories

(Participant/National Laboratory Name)

QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1-2000

This Deliverable was subjected to:

☒ Technical Review	☐ Peer Review
Technical Review (TR)	**Peer Review (PR)**
Review Documentation Provided	**Review Documentation Provided**
☐ Signed TR Report or,	☐ Signed PR Report or,
☐ Signed TR Concurrence Sheet or,	☐ Signed PR Concurrence Sheet or,
☒ Signature of TR Reviewer(s) below	☐ Signature of PR Reviewer(s) below

Name and Signature of Reviewers

Paul Mariner	
Bill Arnold	

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may be also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.

Acknowledgements

This work was supported by the U.S. Department of Energy (DOE) Office of Nuclear Energy within the Office of Used Nuclear Fuel Disposition Research and Development.

The independent peer reviewers, Abe van Luik and Martha Pendleton for Revision 0, and Paul Mariner and Bill Arnold for Revision 1, provided comments that greatly improved the quality of the document. Helpful comments were also provided by DOE reviewers Mark Tynan and coworkers from the Office of Used Nuclear Fuel Disposition Research and Development.

This report is submitted for Milestone M2FT-13SN0808041 in Work Package FT-13SN080804.

CONTENTS

CONTENTS (cont.)

CONTENTS (cont.)

CONTENTS (cont.)

FIGURES

FIGURES (cont.)

FIGURES (cont.)

TABLES

TABLES (cont.)

ACRONYMS

Andra	French National Radioactive Waste Management Agency
BMWi	Federal Ministry of Economics and Technology
BSC	Bechtel SAIC Company
BWR	boiling water reactor
CFR	Code of Federal Regulations
DRZ	disturbed rock zone
EBS	engineered barrier system
EC	European Commission
EDZ	excavation disturbed zone
EPRI	Electric Power Research Institute
ERB	example reference biosphere
FEPs	features, events, and processes
FY	fiscal year
GDS	generic disposal system
HLW	high-level radioactive waste
IAEA	International Atomic Energy Agency
ICRP	International Committee on Radiation Protection
MTHM	metric tons heavy metal
MTU	metric tons uranium
Nagra	Swiss National Cooperative for the Disposal of Radioactive Waste
NBS	natural barrier system
NEA	Nuclear Energy Agency
ONDRAF/NIRAS	Belgian Agency for Radioactive Waste and Enriched Fissile Materials
PA	performance assessment
PAMINA	Performance Assessment Methodologies in Application to Guide the Development of the Safety Case
PWR	pressurized water reactor
SAFIR	Safety Assessment and Feasibility Interim Report

SCK•CEN	Belgian Nuclear Research Centre
SKB	Swedish Nuclear Fuel and Waste Management Company
SKBF/KBS	Swedish Nuclear Fuel Supply Company/Division KBS
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
THCMBR	thermal, hydrologic, chemical, mechanical, biological, and radiological
TRU	Transuranic
U.S. DOE	U.S. Department of Energy
U.S. EPA	U.S. Environmental Protection Agency
U.S. NRC	U.S. Nuclear Regulatory Commission
UFD	Used Fuel Disposition
UNF	used nuclear fuel
UREX	uranium extraction
WIPP	Waste Isolation Pilot Plant
WP	waste package

USED FUEL DISPOSITION CAMPAIGN GENERIC DEEP GEOLOGICAL DISPOSAL SAFETY CASE

1 PURPOSE OF THE GENERIC DEEP GEOLOGIC DISPOSAL SAFETY CASE

1.1 Introduction

This Generic Deep Geologic Disposal Safety Case presents generic information that is of use in understanding potential deep geologic disposal options in the United States for used nuclear fuel (UNF) from reactors and high-level radioactive waste (HLW) resulting from the reprocessing of used nuclear fuel. Potential disposal options include mined disposal in a variety of geologic media, and deep borehole disposal in basement rock. The intent of the document is not to provide a basis for comparison of these alternatives or to support selection of a specific site and/or disposal option for development of a geologic disposal facility. Rather, the Generic Safety Case is intended to be a source of information to provide answers to questions that may arise as the U.S. works to develop strategies to dispose of used nuclear fuel and high-level radioactive waste.

Used nuclear fuel is the terminology used to describe irradiated fuel withdrawn from a nuclear reactor and stored pending reprocessing, recycling, or for which the manner of disposition has not been determined. Spent nuclear fuel (SNF) refers to irradiated fuel withdrawn from a nuclear reactor that is intended for permanent disposal without further reuse. In the U.S., used nuclear fuel (or used fuel) is the preferred terminology in recognition of the fact that future strategies may include reprocessing steps before disposal decisions are made, and is used extensively in this safety case. In non-U.S. programs, the distinction between used nuclear fuel and spent nuclear fuel is not made, and spent nuclear fuel (or spent fuel) is used to refer to all irradiated fuel from reactors. In this safety case, the term spent nuclear fuel is preserved where necessary to retain its original meaning. The term high-activity waste refers collectively to both used nuclear fuel and high-level radioactive waste (U.S. Nuclear Waste Technical Review Board 2011).

This Generic Safety Case document anticipates a need for documentation that will serve to inform stakeholders, decision makers, and regulators facing each of the number of decisions that will need to be made as the country restructures its high-activity waste disposal program. This Generic Safety Case is patterned after the structure of the Nuclear Energy Agency safety case consensus document (Organisation for Economic Co-operation and Development 2004). It is consistent with new initiatives of the Nuclear Energy Agency for methods of safety assessments for geologic disposal facilities (Organisation for Economic Co-operation and Development 2012). The safety case approach is outlined in Section 1.2.

The *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management* (International Atomic Energy Agency 1997) defined the disposal of used nuclear fuel as “the emplacement of spent fuel or radioactive waste in an appropriate facility without the intention of retrieval.” The Joint Convention agreements include among other items, “to provide for effective protection of individuals, society and the environment, by applying at the national level suitable protective methods as approved by the regulatory body, in the framework of its national legislation which has due regard to internationally endorsed criteria and standards; to strive to avoid actions that impose reasonably predictable impacts on future generations greater than those permitted for the current generation; and, to aim to avoid imposing undue burdens on future generations.”

Further, a recent directive (EURATOM 2011) establishing a community framework for the responsible and safe management of used fuel and radioactive waste, noted that, “…. Safety decisions should be based on the findings of an assessment of safety and information on the robustness and reliability of that assessment and the assumptions made therein. The decision-making process should therefore be based on a collection of arguments and evidence that seek to demonstrate that the required standard of safety is achieved for a facility or activity related to the management of spent fuel and radioactive waste. In the particular case of a disposal facility, the documentation should improve understanding of those aspects influencing the safety of the disposal system, including natural (geological) and engineered barriers, and the expected development of the disposal system over time."

For the past half century, the U.S. has undertaken efforts to develop mined geologic disposal facilities to address the ever increasing volumes of high-activity wastes in the country created by the accumulation of used nuclear fuel and radioactive waste from reprocessing and other sources. The U. S. Department of Energy is revisiting the investigation of a variety of geologic media and concepts for the disposal of the used nuclear fuel and high-level radioactive waste that exist today, and that could be generated under future fuel cycles.

Disposal of high-activity waste in a range of geologic media has been investigated by other countries as well. These long-lived radioactive wastes must be appropriately contained, and isolated from humans and the environment for many thousands of years. Considerable progress has been made in the U.S. and other nations, and has resulted in an international consensus that "our responsibilities to future generations are better discharged by a strategy of final disposal than by reliance on stores which require surveillance, bequeath long-term responsibilities of care, and may in due course be neglected by future societies whose structural stability should not be presumed" (Organisation for Economic Co-operation and Development 1995). Mined geological disposal is the currently favored radioactive waste management approach providing long-term security and safety in a manner that does not require active monitoring, maintenance, and institutional controls. Staged development of a repository will allow for protecting the interests of future generations.

A National Research Council committee (National Academy of Sciences 2001) noted that the growing inventory of high-activity waste requires attention by national decision makers; safety and security are being achieved by storage, often at or near the facility that produced the waste; and the inventories are increasing beyond the capacity that can be stored in existing facilities. The committee noted that measures must be taken to deal with this, and the feasible options are monitored storage on or near the earth's surface and geological disposal. The committee observed that geological disposal remains the only scientifically and technically credible long-term solution available to meet the need for safety without reliance on active management; and that geologic disposal would place fissile materials out of reach of all but the most sophisticated weapons builders. They noted that while providing convincing evidence of long-term safety of any repository is a technical challenge, a well-designed repository represents, after closure, a passive system containing a succession of robust safety barriers, and that our present civilization designs, builds, and lives with technological facilities of much greater complexity and higher hazard potential.

The international Radioactive Waste Management Committee (Organisation for Economic Co-operation and Development 2008) concluded that "[d]isposal can be accommodated in a broad range of geological settings as long as these settings are carefully selected and matched with appropriate facility design and configuration and engineered barriers" and "[t]he overwhelming scientific consensus worldwide is that geological disposal is technically feasible". This is supported by the extensive experimental data accumulated for different geological formations and engineered materials from surface investigations, underground research facilities, and demonstration equipment and facilities; by the current state of the art in modeling techniques; by the experience in operating underground repositories for other classes of wastes; and by the advances in best practice for performing safety assessments of potential disposal systems." Finally, the Committee noted that "[a] geological disposal system provides a unique level and duration of protection for high-activity, long-lived radioactive wastes. The concept takes advantage of the capabilities of both the local geology and the engineered materials to fulfill specific safety functions in complementary fashion, providing multiple and diverse barrier roles."

The U.S. Department of Energy Office of Nuclear Energy, Office of Fuel Cycle Technology has established a Used Fuel Disposition (UFD) Campaign to conduct the research and development activities related to storage, transportation, and disposal of used nuclear fuel and high -level nuclear waste. The mission of the Used Fuel Disposition Campaign is (U.S. Department of Energy 2012):

> *To identify alternatives and conduct scientific research and technology development to enable storage, transportation and disposal of used nuclear fuel and wastes generated by existing and future nuclear fuel cycles.*

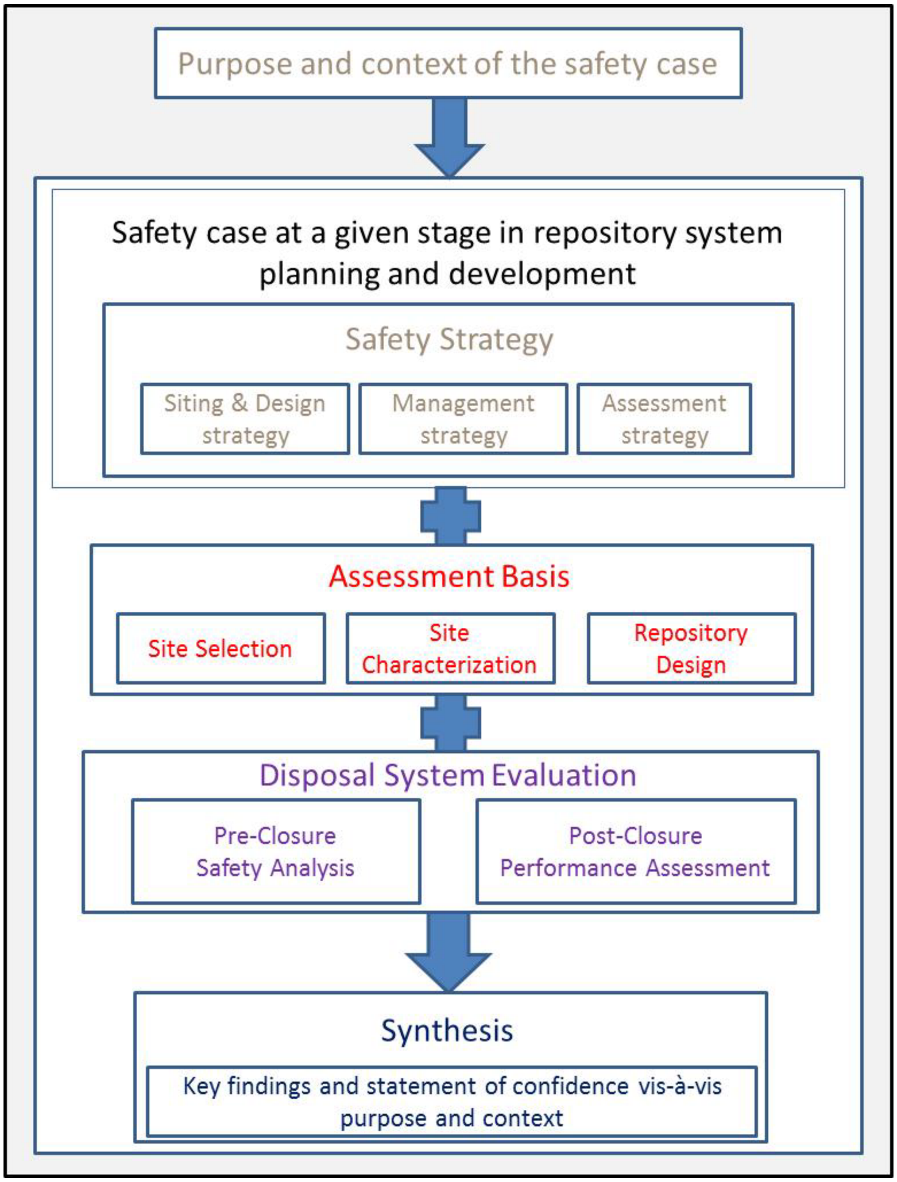
The work undertaken builds on past work in these areas both in the U.S. and in other countries. The U.S. national laboratories have participated in these programs and have conducted research related to these technologies; however, comprehensive programs investigating a variety of geologic media and disposal options have not been a part of the U.S. waste management program since the mid-1980s. Such a comprehensive research and development program is being developed and executed within the Used Fuel Disposition Campaign. In the initial stages, the program is examining combinations of generic geologic media and facility designs that could potentially support development of a geologic disposal facility.

1.2 The Safety Case Approach

A widely accepted approach for documenting the basis for the understanding of the disposal system, describing the key justifications for its safety, and acknowledging the unresolved uncertainties and of their safety significance is a document known as a safety case (Organisation for Economic Co-operation and Development 2004; International Atomic Energy Agency 2006). The safety case is developed to support all aspects of development of the disposal concept and elucidates the approaches for the management of issues related to such development. This provides a basis for making decisions relating to the development, operation, and closure of the facility, and allows attention be focused on areas where further understanding of those aspects influencing the safety of the geological disposal facility is needed. The development of some form of a safety case and supporting safety assessments for review by the regulator and other interested parties is central to the development, operation, and closure of a geological disposal facility.

The development of a geologic disposal facility, including siting, design, construction, operation, and closure, is likely to take place over several decades. In most countries, plans for repository development envision the disposal facility being developed in a series of steps. The safety case serves an important role in informing stakeholders about the progress being made as these steps proceed. The steps involve decisions about identifying sites as possible candidates, screening against well-defined criteria, performing site characterization studies on those sites selected for further evaluation, recommending a site for development as a repository, participating in the licensing proceedings for the repository facility, and the construction, operation, closure, and decommissioning of the facility. Each of these steps involves, in an iterative manner: the accumulation and assessment of necessary data; the development of disposal concepts; studies for design and safety assessments with progressively improving data; reviews; public consultations; and eventual decisions. The safety case matures with the evolution of the program, and helps support transparency and provides information to all stakeholders.

The step-by-step approach, together with the consideration of a range of options for the disposal facility, is expected to be responsive to new information and advances in technologies; address concerns of decision makers and stakeholders; and preserve the option of retrieving the waste after its emplacement if deemed appropriate. Figure 1-1 provides an overview of the components of a safety case for geologic disposal.



Source: Freeze et al. 2012, Figure 1-1; modified from Organisation for Economic Co-operation and Development 2004, Figure 1.

Figure 1-1. Elements of a Safety Case for Geologic Disposal

At the heart of a safety case is the synthesis of evidence, analyses, and arguments that quantify and substantiate a claim that a repository will be safe after closure and the time of reliance on active control and monitoring of the facility. The safety case becomes more comprehensive and rigorous as a program progresses, and can support decision making at several steps in the repository planning and implementation process. A key function of the safety case is to provide a platform for informed discussion whereby interested parties can assess their own levels of confidence in a project, determine any reservations they may have about the project at a given planning and development stage, and identify the issues that may be a cause for concern or on which further work may be required. Safety assessments are carried out periodically, and are used to develop and progressively update the safety case.

A safety assessment is an analysis to predict the long-term performance of the overall system and its impact and confidence in the assessment of safety, where the performance measure is radiological impact or some other global measure of impact on safety (Organisation for Economic Co-operation and Development 1999b; Organisation for Economic Co-operation and Development 2004). A safety assessment addresses the ability of a site and repository facility design to meet the applicable technical requirements and provide for the safety functions. Safety assessment includes quantification of the overall level of performance, analysis of the associated uncertainties and comparison with the relevant design requirements and safety standards. As site investigations progress, safety assessments become increasingly refined, and, at the end of a site investigation, sufficient data should be available to support a safety assessment to demonstrate compliance with regulatory safety standards. Safety assessments also identify any significant deficiencies in scientific understanding, data, or analysis that might affect the results presented. Depending on the stage of development, safety assessments may be used to aid in focusing research and/or to assess compliance with the various safety objectives and standards (International Atomic Energy Agency 2006).

To support the ultimate goal of demonstration of compliance with regulatory safety standards, the safety assessments must evolve to the point wherein site characterization studies have led to quantification of uncertainties, the system performance assessment models are well developed, and the facility designs sufficiently complete to allow analysis. Nonetheless, there is considerable value in early safety assessments performed using generic disposal system models. These models can take advantage of developmental work undertaken in earlier U.S. studies, and of that currently being undertaken in other geologic disposal investigations being conducted world-wide. Knowledge of parameters that are important to containment and isolation leads to the definition of important barriers to radionuclide movement, which in turn helps focus and refine safety assessment models, facility designs, and site investigation programs.

As the safety case is developed, the safety strategy, which is the high-level approach adopted for achieving safe disposal, can also be described. The safety strategy depends heavily on the standards and criteria that will be used to assess the overall safety of the geologic disposal facility. The safety strategy includes the strategies for the overall management of the various activities required for geologic disposal facility planning and implementation, for siting and design, and for performing safety assessments. The safety strategy should be aligned with the requirements of the project, capable of achieving project goals, and tackling future decisions. The safety strategy is discussed further in Section 2.

At early stages in the deployment of geologic disposal facilities (e.g., the early phases of site identification and screening) it is not possible to have detailed site specific information, designs, and models. Yet it is at this point where technical information related to performance, even if it is not fully developed, should be available to stakeholders and decision makers. By preparing a document that compiles information important to stakeholders and decision makers, and updating it as the geologic disposal program matures, the Used Fuel Disposition Campaign can provide readily-available information supporting future decisions pertaining to the disposal of used nuclear fuel and high-level radioactive waste in the U.S.

While the safety assessment is the principal technical basis for determining the importance of system elements, it is not sufficient. The safety case substantiates the safety, and contributes to confidence in the safety, of the geological disposal facility. The safety case is an essential communication tool for decision-making process concerning the development of a geologic facility. It includes the output of safety assessments, together with additional information, including supporting evidence and reasoning on the robustness and reliability of the facility, its design, the design logic, and the quality of safety assessments and underlying assumptions. The safety case may also include more general justifications relating to the need for the disposal of radioactive waste, and information to put the results of the safety assessments into perspective. Further, it aids in addressing perceptions of safety through the incorporation of supporting information.

Issues determined not to be important to either performance (safety assessment) or the design/construction of the disposal system still may be of importance to build confidence in the overall safety case. As an example, issues associated with features that may not be important to performance in the safety analysis, but act as part of a multiple-barrier system that demonstrate defense in depth could be of importance with respect to confidence in the overall safety case.

1.3 Objectives of the Generic Safety Case

A path forward for the U.S. high-activity waste disposal program will need to consider a number of possible alternatives. Information presented in this Generic Safety Case may support deliberations about the role of geologic disposal in the U.S. nuclear fuel cycle and initial site screening. The information presented in this Generic Safety Case will be available to provide decision makers and stakeholders a concise summary of technical information that may be germane to their deliberations. Significant decision points early in a disposal program could also include the definition of the types and amount of waste to be disposed of, the choice of potential host rock types, regions of the country where likely candidate sites could be located, and potential engineering concepts.

There is ample precedent in the U.S. for development of documents that serve similar purposes. The U.S. Department of Energy (2008b) and preceding documents (U.S. Department of Energy 1998; U.S. Department of Energy 2001; U.S. Department of Energy 2002b; U.S. Department of Energy 2007) presented the type of information to be expected in a safety case document. These documents were developed to provide comprehensive information to stakeholders or decision makers at points in time when important decisions needed to be made about the future course of the U.S. high-activity waste disposal program. These documents were augmented by other publically-available information, such as semi-annual reports and evaluations of review and oversight groups, where technical information was reviewed. There was, however, no single integrated source of updated information presented in a user-friendly format for stakeholders and decision makers. By developing this Generic Safety Case before the siting process begins, the U.S. Department of Energy is attempting to provide ready access to information for stakeholders, and a potential mechanism to assist in the identification of stakeholder issues.

Planning and implementation of the development of a geologic disposal facility are expected to occur in a stepwise manner over several decades, punctuated by decision points. Information, such as that compiled in a safety case, should be sufficiently detailed and comprehensive at each decision point to provide the necessary technical input for informing decisions necessary at that point to move to the next phase of the program. If the U.S. elects to proceed with identifying or evaluating candidate sites for a geologic disposal facility, one of the first decision points would be examining and evaluating the information concerning candidate sites. The Generic Safety Case provides preliminary evaluations of the safety of potential geologic disposal facilities, geologic media, and disposal technologies that could be considered as the country moves forward. Four specific disposal options currently under consideration by the Used Fuel Disposition Campaign are described in Section 1.4.

A key function of this initial iteration of the Generic Safety Case is to compile relevant information and guide the activities of the Used Fuel Disposition Campaign. Once a site is identified and an initial engineering concept defined, the decisions may involve more detailed planning including the scope of above- and below-ground investigations, demonstrations of the engineering feasibility of key elements, choices between design variants, and the refinement of the underground layout. The safety case will be developed progressively as the project proceeds, and can be presented at each key step in the development of the geological disposal facility. This initial iteration of the safety case provides a mechanism to communicate the understanding of safety to the broader audience of stakeholders and may reduce discrepancies between the understanding and expectations of the different stakeholders. Issues identified by others may mandate an update or revision of the safety case before moving forward. The formality and level of technical detail of the safety case depend on the stage of development of the project, the decision at hand, and the audience to which it is addressed.

The emphasis of this Generic Safety Case is on the long-term postclosure safety of the repository or disposal system. It is an integration of justifications and evidence that describe, quantify, and substantiate the safety, and the level of confidence in the safety, of the geological disposal facility. Key postclosure safety elements to be addressed in the evolution of this Generic Safety Case include:

- Quantitative assessments of long-term performance
- Process for selection of a site and design of a repository that provide for defense in depth, a system of multiple features designed to ensure that failure of one feature does not result in failure of the entire system. The system would also provide a margin of safety against radionuclide releases and a margin of safety compared to applicable radiation protection standards.
- Qualitative insights gained from the study of natural and man-made analogues to the repository or to processes that may affect system performance
- Management and monitoring considerations to ensure the integrity and security of the repository (e.g., a performance confirmation program) and enable sound scientific and engineering bases for later decisions.

Current research and development needs were prioritized to ensure that they address generic issues and data needs, and were important to developing the initial safety case and safety assessments for the four disposal options outlined in Section 1.4. These current research needs and priorities and their technical bases are described in the *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) and are discussed further in Section 5.3.1.

As information is obtained (e.g., from research and development activities, generic studies, and generic safety assessments), designs mature, and decisions are made, the Generic Safety Case document can be updated to document the progress that has been made, the issues that remain to be solved, and the status of technical evaluations supporting the decisions to move from one phase to the next.

1.4 Disposal Options Considered

The Used Fuel Disposition Campaign is currently evaluating the viability of four geologic disposal options: mined repositories in salt, clay, and granite, and deep borehole disposal in crystalline rock (Rechard et al. 2011). For each of these disposal options, the rock type is identified at a broad level:

- **Salt**—Refers to both bedded and domal evaporitic formations. Bedded salt formations typically consist of thick layers of relatively pure halite (sodium chloride) interspersed with thinner layers of materials such as anhydrite, shale, dolomite, and other salts such as potassium chloride (Hansen and Leigh 2011). In this report, a mined repository in this rock type is referred to as the salt disposal option.
- **Clay**—Refers to a broad range of fine-grained, detrital sediments ranging from poorly unconsolidated clays to lightly indurated argillaceous media, including mudstone, claystone and soft clays, and shale. In this report, a mined repository in this rock type is typically referred to as the clay disposal option, although, in few cases, it is referred to as a clay/shale or shale disposal option. Argillite, a compact rock that has undergone a somewhat higher degree of induration than mudstone or shale and is less clearly laminated, is also included in this rock type.
- **Granite**—Refers to a range of igneous and metamorphic lithologies, including granite and other granitic rocks, and high-grade crystalline metamorphic rocks. In this report, a mined repository in this rock type is typically referred to as the granite disposal option, although, in few cases, it is referred to as a crystalline rock disposal option.
- **Crystalline rock**—Refers to large bodies of igneous or metamorphic rock, similar to the granite rock type above. In this report, crystalline rock is most commonly used to refer to deep basement rock considered for the deep borehole disposal option.

While these four disposal options are used as the bases for evaluation in this Generic Safety Case, they are not presented as a final list of the best possible alternatives, and it is recognized that other options have been identified in the past, or may be identified in the future, that also have the potential to provide safe long-term isolation. There are several reasons for focusing on these four disposal options at this stage of the program.

First, the U.S. went through an extensive review of all available options for disposal and management during the 1970s, culminating in the *Environmental Impact Statement on Management and Disposal of Commercially Generated Radioactive Wastes* (U.S. Department of Energy 1980). This review considered a full range of alternatives to mined geologic repositories, including deep boreholes, subseabed disposal, space disposal, and ice sheet disposal. Mined repositories were the favored option, but subseabed disposal and deep boreholes were retained for further consideration (U.S. Department of Energy 1981). Ocean disposal was precluded by international treaty (International Atomic Energy Agency 1999a) in the 1990s. Deep boreholes were considered to require further technological advances, and disposal programs in both the U.S. and other nations focused on mined repositories beginning in early 1970s. The U.S. program evaluated salt, shale, granite, basalt, and volcanic tuff (U.S. Department of Energy 1986g) before focusing exclusively on volcanic tuff.

Second, conclusions drawn in the U.S. program in the early and mid-1980s about the potential viability of salt, clay, and granite as disposal media have been confirmed by extensive work internationally. Salt has been shown to be a viable medium for disposal of non-heat-generating transuranic (TRU) waste at the Waste Isolation Pilot Plant (WIPP) in the U.S., and research in Germany continues to show promise for the disposal of heat-generating waste in salt. Clay disposal concepts have been evaluated in France, Belgium, and Switzerland. Granite (crystalline) repository concepts have been evaluated in Sweden, Finland, Switzerland, and Japan. Other geologic media are under consideration for specific purposes (e.g., Canada is investigating the use of a mined repository in carbonate rock to dispose of intermediate level waste, and the U.S. has disposed of low-level radioactive waste and transuranic waste in near-surface alluvium). These programs are discussed in Appendix C.

Third, deep borehole disposal continues to be the primary viable alternative to mined repositories. Deep borehole disposal was considered in a waste management environmental impact statement (U.S. Department of Energy 1980). The U.S. Department of Energy further investigated the concept in the 1990s for the disposal of surplus plutonium and the Swedish high-level nuclear waste program conducted a feasibility study in 2000. More recently, studies have continued at the University of Sheffield in the United Kingdom, at the Massachusetts Institute of Technology, and at Sandia National Laboratories in the U.S. (Rechard et al. 2011). U.S. studies in 2009 and 2011 proposed additional design details for borehole completions and waste canisters (Brady et al. 2009; Arnold et al. 2011).

Finally, no new information has been developed since the early 1980s to suggest that options evaluated and screened from further consideration at that time (e.g., space disposal or ice-sheet disposal) should be re-evaluated.

1.5 Overview of Document

In this Generic Safety Case a number of important qualitative issues regarding the safety of potential geologic disposal systems in a variety of geologic media are addressed; the salient results of generic quantitative assessments of long-term repository performance are also presented to provide information about the performance and safety aspects of four geologic disposal options.

Section 2 presents information about the three elements of the safety strategy: the management strategy, the siting and design strategy, and the assessment strategy. The management strategy discussion includes examining evidence for the soundness of geological disposal as a waste management option, such as the reduced likelihood and consequences of inadvertent human intrusion and the reduced likelihood of terrorist attacks on stored radioactive wastes. It also discusses another attribute of geologic disposal—the deliberately cautious approach to disposal facility development that provides multiple opportunities for stakeholder involvement—and includes a summary of the multiple qualitative justifications for long-term safety of a geologic disposal facility. The siting and design strategy discussion includes a discussion of natural barriers, a summary of studies of available area in the U.S. with the potential for siting geologic disposal facilities, and a discussion of engineered barriers and potential repository concepts. The assessment strategy discussion presents the safety assessment methodology, and also includes a discussion of metrics to assess long-term performance.

Section 3 presents information about the assessment basis. The assessment basis discussion includes an overview of previous safety assessments of the performance of several geologic disposal systems followed by a summary of the assessment basis for four generic disposal systems considered in this safety case. The generic assessment basis includes potential inventory characteristics, natural barrier characteristics, and engineered barrier characteristics.

Section 4 describes the development and application of postclosure safety assessment models to assess the long-term performance of four generic geologic disposal systems. The discussion of model development includes feature, event, and process (FEP) analysis, scenario development, and the construction of conceptual, mathematical, and computational models. The model application includes deterministic results from selected scenarios and sensitivity analyses.

Section 5 addresses safety assurances for the future, including confidence in the face of uncertainty. Elements of this confidence include: the fact that additional data will be collected to support the siting evaluations; increased oversight throughout the process; and regulatory requirements to assure monitoring and physical protection into the distant future. Section 5 also addresses readiness for moving from the generic evaluations presented in this Generic Safety Case to a site screening phase.

This document combines relevant initial work of the Used Fuel Disposition Campaign, using the expertise of U.S. national laboratory scientists, and information from geologic disposal programs in other countries, to present an evaluation of the likelihood that geologic media and disposal technologies available in the U.S. may provide safe repositories for used nuclear fuel and high-level radioactive waste, whether from past reprocessing or future advanced fuel cycles. The focus of this initial iteration of a Generic Safety Case is the postclosure safety of potential generic geologic disposal systems. Information about preclosure transportation safety and operational safety, which is also relevant to a safety case, is deferred to a later iteration.

2 SAFETY STRATEGY FOR GENERIC DEEP GEOLOGIC DISPOSAL SYSTEMS

A safety case is the synthesis of evidence, analyses, including safety assessments, and arguments that quantify and substantiate a claim that a geologic disposal facility will be safe. The work presented in this Generic Safety Case supports a conclusion that the U.S. has multiple, viable, options for developing a geologic disposal facility in the future. The safety case approach described in Section 1 is a step-by-step process that is expected to be responsive to new information and advances in technologies and address various challenges as the design of a geologic disposal facility matures toward development of the facility. Figure 1-1 provides an overview of the components of a safety case for geologic disposal. The safety case includes a safety strategy that has three components: a management strategy, a siting and design strategy, and an assessment strategy.

The safety strategy is the high-level approach adopted for achieving safe disposal (Organisation for Economic Co-operation and Development 2004). The safety strategy should encompass the requirements of the project, be capable of achieving project goals, and be focused on obtaining the information needed to support future decisions.

A management strategy incorporates good management and engineering principles and practice, including maintaining sufficient flexibility within a step-wise planning and implementation process, to address unexpected site features, technical difficulties, or uncertainties that may be encountered. This includes the strategies for the overall management of the various activities required for repository planning and implementation, for siting and design, and for performing safety assessments. It also should seek to be aware of and take advantage of advances in scientific understanding and engineering techniques. This management function keeps work focused on project goals, allocates resources to particular activities, and ensures that these activities are correctly carried out and coordinated.

The siting and design strategy should generally be based on principles that favor robustness in natural and engineered components that could be important to waste isolation, and limit uncertainty, including through the use of a multiple barrier concept. The siting and design strategy should seek to select a site that is expected to meet applicable requirements, and develop practicable engineering solutions, consistent with the characteristics of the selected site and the waste forms to be disposed.

The assessment strategy must ensure that safety assessments capture, describe and analyze uncertainties that are relevant to safety, and investigate their effects. The assessment strategy should endeavor to define the approach to evaluate evidence, perform safety assessments, and analyze the evolution of the system. This information can be used to develop or update the safety case. These components are closely connected in that a sound management strategy and a well sited and designed system will facilitate the development of a competent and convincing safety case. All are required, however, and shortcomings in any one cannot be overcome by excellence in the others.

2.1 Management Strategy

2.1.1 Overview

The safety strategy is a management tool to create and direct a program that is capable of complying with the requirements of the project while achieving the project’s goals. The strategy is focused on developing a safety case and therefore must include the overall management of the various activities required, which include activities for siting and design and for performing safety assessments. The management strategy builds from sound management and engineering principles, which include quality assurance, records management, and cost and schedule tracking. In order to oversee the development of the safety case, management attends to concerns raised at each stage of the development of the safety case, and ensures that any unexpected features of the geologic disposal facility system that are discovered as the development progresses are addressed.

Producing a safety case, however, involves more than selecting a site, developing a design, and assessing its safety. The safety case includes the synthesis of all evidence that provides confidence in the decisions to move forward at each stage of facility development. The safety case should become more comprehensive through time and support decision making. Because it can provide a platform for informed discussion among interested parties with differing concerns and levels of interest, it is incumbent on management to provide information so that individuals can assess their own levels of confidence in a project, determine any reservations they may have about the project at a given planning and development stage, and identify the issues that may be a cause for concern. The management strategy functions that ensure the development of the safety case must build on the inherent robustness of geologic disposal.

Management strategies must address the possibility of inadvertent human intrusion. Awareness of the potential for human intrusion could result in management actions to limit or avoid the possibility. Management strategies must also address terrorism concerns; while underground emplacement would appear to be inherently safe, nonetheless, this is often a concern of stakeholders. Vulnerabilities of repository surface facilities must also be addressed by the strategy. Consideration of analogues that exhibit responses to the types of processes that a geologic disposal facility is likely to experience is often useful in developing confidence in reliance on natural and engineered barriers. There is value if repository features or components can be shown to be robust by demonstrating that an analogous natural feature is resistant to a process to which a geologic disposal facility is likely to be exposed. The qualitative or less quantitative evidence for safety from natural analogues may be more accessible, more convincing, and more familiar to the public than the results of complex mathematical models (Organisation for Economic Co-Operation and Development 2004).

A management strategy that is cautious is also of value in developing a safety case. Confidence on the part of stakeholders can be enhanced if there is a sense that development of a geologic disposal facility and its safety case is able to address concerns as they are raised. If the development proceeds faster than stakeholders are able to raise concerns and have them addressed, there is a potential for loss of confidence in the program. Finally, a management strategy that builds on multiple qualitative aspects of the rationale for long-term safety should reinforce and add to confidence in the siting and design activities as well as the safety assessments.

The following sections present additional information on these elements of a management strategy:

- A description of the safety and robustness of geologic disposal as a waste management option (Section 2.1.2.1)
- Addressing inadvertent human intrusion (Section 2.1.2.2)
- Addressing terrorism and radioactive wastes (Section 2.1.2.3)
- The value of analogue evidence for the robustness of geologic disposal (Section 2.1.2.4)
- The value of a staged and cautious facility implementation process (Section 2.1.2.5)
- The existence of multiple aspects of the qualitative rationale for long-term safety (Section 2.1.2.6)

2.1.2 Elements of a Management Strategy

2.1.2.1 The Safety and Robustness of Geologic Disposal

There is a long acknowledged need to provide a safe approach to dealing with the used nuclear fuel and high-level radioactive waste resulting from commercial power generation and defense activities. Multiple in-depth reviews from the late1950s to the present have noted the need for geologic disposal: “Geological disposal remains the only long-term solution available” (National Academy of Sciences 2001); and “Every nation that is developing disposal capacity plans to use a deep, mined geologic repository for this purpose. Other disposal options (e.g., deep boreholes) have been considered and may hold promise in the long term but are at a much earlier stage of development." (Blue Ribbon Commission on America's Nuclear Future 2012).

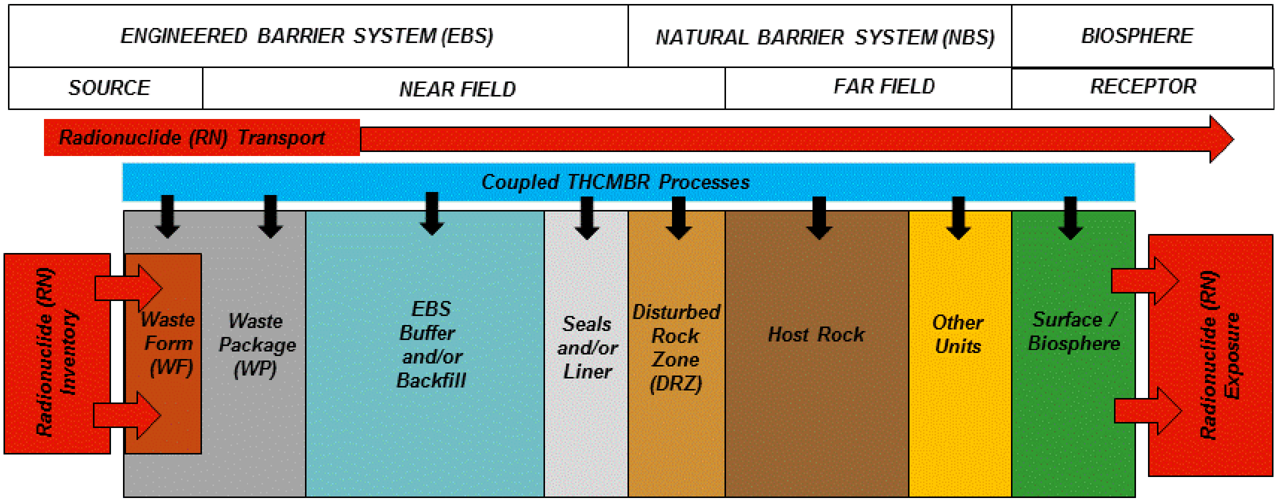
The U.S. Department of Energy (1980) examined a wide range of potential options for disposal of used nuclear fuel and high-level radioactive waste and adopted a strategy to develop mined geologic repositories while maintaining very deep borehole disposal as a potential backup technology (U.S. Department of Energy 1981). The Blue Ribbon Commission on America's Nuclear Future (2012) noted:

> *Deep geologic disposal capacity is an essential component of a comprehensive nuclear waste management system for the simple reason that very long-term isolation from the environment is the only responsible way to manage nuclear materials with a low probability of re-use, including defense and commercial reprocessing wastes and many forms of spent fuel currently in government hands. The conclusion that disposal is needed and that deep geologic disposal is the scientifically preferred approach has been reached by every expert panel that has looked at the issue and by every other country that is pursuing a nuclear waste management program.*

That conclusion is consistent with previous studies in the U.S. concerning approaches to dealing with the wastes for nuclear power generation and defense related activities. Appendix A presents an overview of the history of repository siting in the U.S., and includes a description of previous assessments of approaches to dealing with used nuclear fuel and high-level radioactive waste.

The Used Fuel Disposition Campaign is tasked to identify and research the generic sources of uncertainty that will challenge the viability of various disposal options, to increase confidence in the robustness of generic disposal concepts in anticipation of site-specific complexity, and to develop the science and engineering tools required to address these goals (Swift 2011).

A geologic disposal facility at its simplest consists of an engineered barrier system (EBS), a natural barrier system (NBS) or geosphere, and a biosphere (Figure 2-1). The engineered barrier system comprises the waste form and waste package, and the engineered features of the geologic disposal facility, typically consisting of buffer material, backfill, excavation liner, and/or seals. The natural barrier system, or geosphere, consists of the host rock, within which the geologic disposal facility is developed, and the other geologic units surrounding it. The biosphere is where the potential receptor resides. The biosphere consists of the surface, which defines the receptor and the receptor's lifestyle, and the characteristics of the environment where the receptor resides. Figure 2-1 also schematically illustrates the phenomena that can affect each of these regions or domains. These phenomena include, at a high level, the coupled thermal, hydrologic, chemical, mechanical, biological, and radiological processes that describe (1) waste form degradation and the source term, (2) radionuclide transport through the engineered barriers, (3) radionuclide transport through the natural barriers (i.e., the geosphere), and (4) radionuclide transport, uptake, and health effects in the biosphere.



NOTE: THCMBR = thermal, hydrologic, chemical, mechanical, biological, and radiological

Figure 2-1. Components of a Generic Disposal System

Figure 2-1 also indicates the near field and far field, which are also commonly used to describe the physical domains of a disposal system. The near field encompasses the engineered barriers and the disturbed rock zone (DRZ). The DRZ is the portion of the host rock adjacent to the engineered barriers that experiences durable (but not necessarily permanent) changes due to the presence of the repository (e.g., hydro-mechanical alteration due to tunnel excavation, thermal-chemical alteration due to waste emplacement). The DRZ is sometimes referred to as the excavation disturbed zone (EDZ). The far field encompasses the remainder of the geosphere and the biosphere.

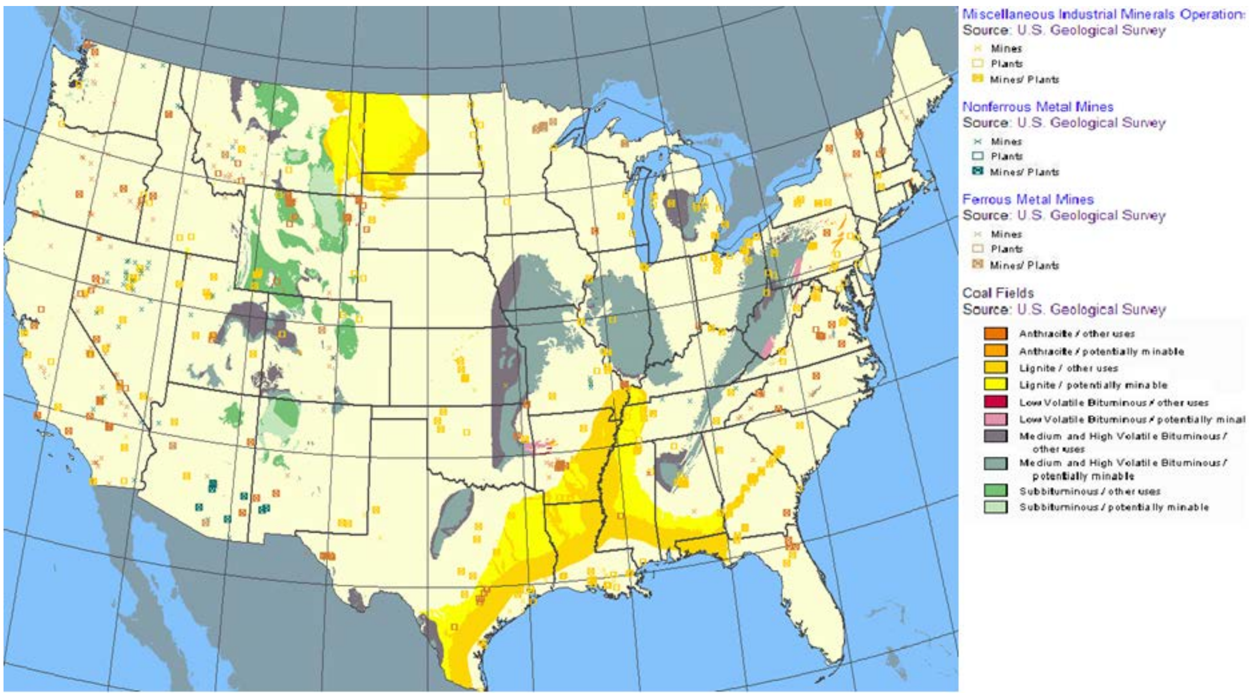
The contiguous 48 states contain many geologic formations that are likely to be technically suitable for deep geologic disposal of nuclear waste. Given appropriate repository designs, there is substantial confidence that reasonable assurance of waste isolation can be demonstrated for several geologic settings, disposal concepts, and rock types, including salt, clay/shale, volcanic rock, granite, and deep boreholes. While no deep geologic disposal facility for high-activity waste has yet been developed to the point of operation, focused characterization programs and geologic disposal facility-relevant data collected from underground research laboratories have not dissuaded experts from the sentiment that geologic disposal constitutes a safe and effective means of long-term waste isolation.

Historically, the U.S. pursued conventional deep geologic repository programs in granite, a subset of igneous or crystalline rocks, clay/shale, salt, and volcanic rock. Granite programs included a full-scale emplacement demonstration in an underground research laboratory at the Climax Stock on the Nevada National Security Site, formerly known as the Nevada Test Site (Patrick 1986). Clay/shale programs were supported by laboratory testing, literature studies, and limited field testing, but no domestic underground research laboratory was developed nor was any disposal demonstration conducted in shale in the U.S. Extensive underground research laboratories, as well as full-scale underground disposal demonstrations were undertaken at several salt sites, including Lyons in Kansas, Avery Island in Louisiana, and Carlsbad in New Mexico.

There have been literature studies, engineering and safety analyses performed for deep borehole disposal, subseabed disposal, and a few deep crustal exploration programs, but none of these disposal options has been demonstrated by the U.S. (Hansen et al. 2011). Other countries have conducted testing in underground facilities in clay, shale, granite, and salt during the last 30 years, with some participation from U.S. researchers, adding to the understanding of generic system performance for those disposal options.

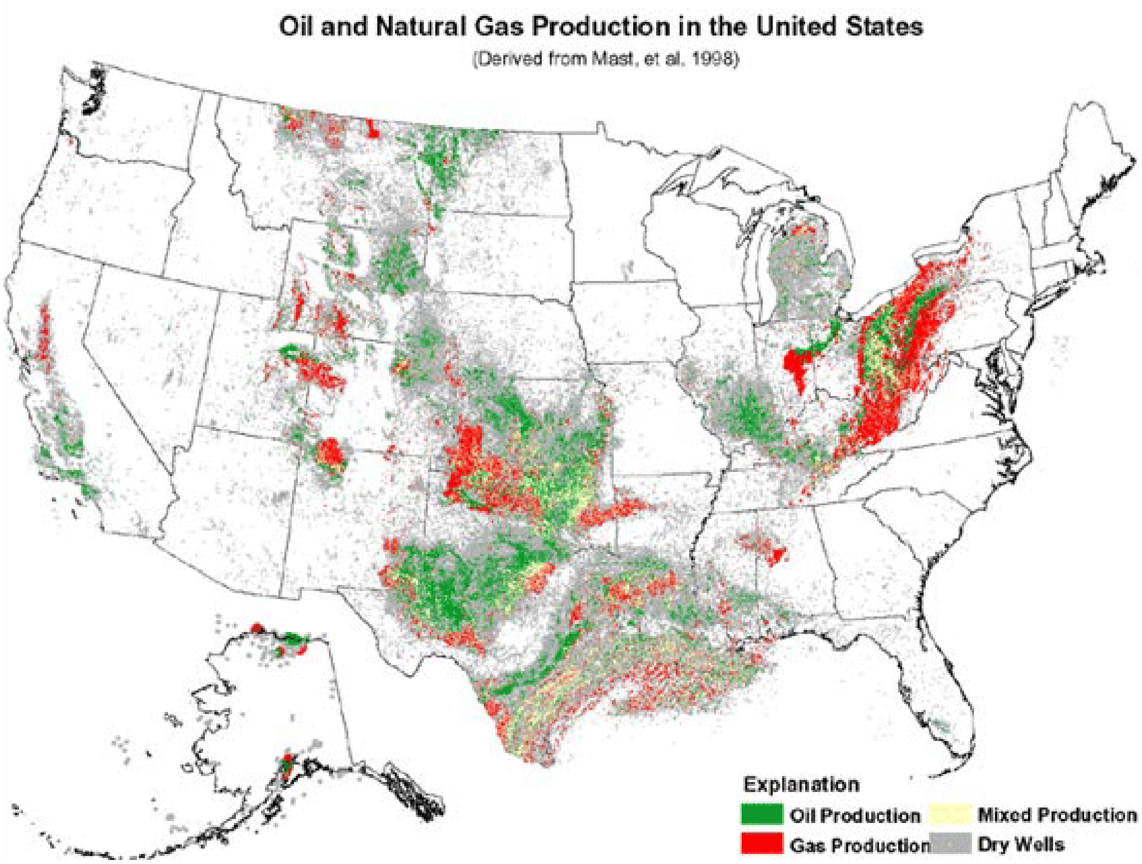
2.1.2.2 Addressing Inadvertent Human Intrusion

One concern that arises in the consideration of the long-term disposal of used nuclear fuel and high-level radioactive waste in a geologic disposal facility is inadvertent human intrusion. As opposed to deliberate intrusion into a closed geologic disposal facility, an inadvertent human intrusion occurs, for example, through drilling exploratory boreholes in search of resources such as water, oil, gas, or minerals. Distribution of ferrous and nonferrous metal mines, industrial minerals, and coal fields of the U.S. is shown in Figure 2-2 (National Atlas 2012). Oil and gas production areas are shown on Figure 2-3 (Tribal Energy and Environmental Information 2012).



Source: National Atlas 2012.

Figure 2-2. Ferrous and Nonferrous Metal Mines, Industrial Minerals, and Coal Fields of the U.S.



Source: Tribal Energy and Environmental Information 2012.

Figure 2-3. Oil and Natural Gas Production in the U.S.

Reduction in the likelihood of an inadvertent human intrusion event may be achieved in a number of ways; for example, by preservation of information, avoidance of resource conflicts, and/or the use of robust waste packages. Measures can be taken to ensure that information regarding the purpose, location, design and contents of the deep geologic disposal system are preserved so that future generations are made aware of the consequences of actions they may choose to take that might affect the disposal system. Avoidance of areas of potential or known resources may be prudent at the screening stage to limit the potential for exploratory drilling in the future. Under certain conditions, robust waste packages can contribute to long-term safety through the mitigation of the degradation of system performance, providing resistance to drilling into the waste package and releasing its contents.

Because closure of a repository would occur many decades in the future, specifics of the proposals for ensuring extremely long-term preservation of information are necessary. International cooperation, including the creation of international archives, would contribute to maintaining knowledge about geologic disposal systems, enhancing long-term safety throughout the world.

The engineered components of the geologic disposal facility should be designed to minimize the likelihood that current-day geologic drilling technology could penetrate the waste. The nature of the wastes themselves also assures that although there is a risk of a substantial dose to a driller encountering a waste package after its degradation and bringing radioactive material to the surface, there is likely to be little consequence for the nearby population. The National Research Council (National Academy of Sciences 1995) provided treatment of this issue that was independent of the type of repository (i.e., for a generic repository). They identified three broad types of hazards from radioactive material that could occur as a result of a repository intrusion of the type characterized by borehole scenarios: hazards to the intruders themselves, hazards that arise because the integrity of the repository's engineered or geologic barriers have been compromised by the intrusion, and hazards to the public from any material brought directly to the surface because such material would be mobile in the biosphere.

The National Research Council (National Academy of Sciences 1995) noted that whenever highly dangerous materials are gathered into one location, an intruder breaking in runs an inevitable risk of being exposed to radiation. All geologic disposal facilities will have this risk, and it is not unique to radioactive waste types, or geologic disposal facility designs, or the geologic media where the facility is located. The National Research Council concluded the following for inadvertent human intrusion:

> *We believe that it would not be feasible to take regulatory actions today to protect the intrusion crew itself against the risks of its actions, except that … active or passive institutional controls might be helpful in this regard.*
>
> *However, it is possible that an inadvertent intruder would not recognize or would irresponsibly ignore the hazard and would leave the cuttings on the surface so that further exposures would occur. …the amount of such future cuttings might not be very different from one repository site or design to another, especially given the unknown nature of an intrusion. Analysis of this hazard too, therefore does not provide information that is useful for judging the ability of the particular repository site and design to protect the public. In this case, we also believe that it is not feasible to take regulatory actions today to alter the repository design to minimize these risks.*

In incorporating the National Academy recommendations into their regulation (40 CFR Part 197), the Environmental Protection Agency accepted the conclusion that analyzing these risks is unlikely to provide useful information about a specific repository site or design; neither did they require that these risks be considered in the compliance analysis for WIPP to address 40 CFR Part 191. It is, however, worth noting that the U.S. Environmental Protection Agency views the frequency and severity of inadvertent human intrusion into geologic disposal facilities differently for different host geosphere media, commensurate with differences in potential resource availability.

2.1.2.3 Addressing Terrorism and Radioactive Wastes

The location of a geologic disposal facility under a significant cover of rock should be sufficient to provide protection against most deliberate human intrusion attempts. After repository operations cease, the subsurface accesses will be sealed, with particular attention paid to preventing deliberate access in the postclosure period. Intuitively, placing the used nuclear fuel deep underground offers protection from terrorist attacks. Reducing the number of physical locations where wastes are gathered allows for more efficient and effective protection against terrorist attacks than does widely distributed storage. For a repository, the cost of protection is associated with construction of the underground emplacement drifts; the security provided is passive, requiring no human intervention after closure.

2.1.2.4 The Value of Analogue Evidence for the Robustness of Geologic Disposal

Because a safety case is intended as a platform for discussions with a broad based audience, including particularly the general public, it may place emphasis on different lines of evidence, rationale, and analyses compared to a safety case aimed at regulators and other technical specialists. The first few hundred years following emplacement of the waste could be the period of highest concern to many members of the public and may be emphasized to a greater degree when safety cases are presented. The qualitative or less quantitative evidence for safety from natural analogues may be more accessible, more convincing, and of more interest to the public than the results of complex mathematical models (Organisation for Economic Co-Operation and Development 2004).

Repository analogues play an important role in understanding how a geologic disposal facility would perform over the very long time periods after it is closed. Natural analogues represent the occurrence of materials or processes similar to those found in, or caused by, a repository. Analogues may provide insights about the robustness of the geologic disposal facility components over timescales that are untestable by scientists studying the behavior of a geologic disposal system. Important management perspectives about the robustness of geologic disposal as a waste management option can be developed from observations of naturally occurring materials and phenomena. There are many naturally occurring materials that are similar to the natural and engineered components, or features, of a geologic disposal facility. Similarly, there are numerous naturally occurring phenomena that are similar to the anticipated changes that the natural and engineered components of a geologic disposal facility are likely to experience over time. In repository development and analysis these phenomena are often referred to as processes. There are also man-made, or anthropogenic, materials that have been exposed to naturally occurring phenomena for a sufficient length of time to have exhibited behavior that is relevant to understanding the long-term behavior of a geologic disposal facility. When studied to understand how a geologic disposal facility might perform over very long periods of time, these materials and phenomena are referred to as analogues. Naturally occurring materials similar to those that would be used in a geologic disposal facility can be found with tens of thousands to millions of years of exposure to the same naturally occurring phenomena that a geologic disposal facility would experience over similar times. This provides scientists an opportunity to make observations about the long-term behavior of the materials comprising a geologic disposal facility and the phenomena to which they are likely to be exposed, and hence, about the long-term safety of the geologic disposal facility itself.

There are analogues for the long-term performance of deep geologic disposal system components in uranium deposits. The main characteristics of, and processes that take place in, their surroundings that are of potential interest as analogues to a geologic disposal system include the composition and long-term performance of uraninite as an analogue of used fuel, the role of redox processes in radionuclide mobilization and retardation, and the influence of colloids and microbial populations on radionuclide mobility. There are analogues that are concerned with the characterization of the long-term stability of bentonite. The main characteristics of and processes that take place that are of potential interest as analogues to a geologic disposal system include the longevity and alteration of bentonite, the function of bentonite as hydrologic barrier and colloid filter, physiochemical caused by heating, and the collapse of the waste package and interaction with other material of the engineered barriers. Hyperalkaline environments are natural occurrences of secondary minerals analogous to those formed during the hydration of Portland cement and result in interstitial waters characterized by very high pH. The study of this type of natural system may be of use in analyzing the safety of a geologic disposal system, particularly with regard to the longevity of cement and its properties, including the speciation and solubility of radionuclides under high pH conditions, and the nature and stability of colloids formed in high pH waters and at the interface between these and neutral waters. Analogues exist for alternate alteration materials that can have hydrochemical characteristics and associated mineral precipitates that are very similar to the conditions that are expected to exist in a disposal facility containing cement.

While anthropogenic materials have existed for much shorter times, and hence have a shorter record of response to the phenomena a repository is likely to experience, they still are of significant importance to understanding the behavior of a geologic disposal facility. The anthropogenic materials are most similar to the engineered barriers of the geologic disposal facility, particularly the metals comprising the waste packages or some waste form material. Knowledge of the degradation (i.e., corrosion) behavior of anthropogenic materials, while not available on the same timescales as that of naturally occurring materials is important because one of the important functions of the waste packages and waste forms is to provide containment when the waste is most intensely radioactive, a period of a few hundred years. Knowledge of the behavior of anthropogenic materials over even relatively short times is meaningful to understanding repository performance.

Anthropogenic materials and artifacts can provide analogue information that is of value to demonstrating the safe long-term performance of repositories. The analogues investigated include the corrosion of cement or metal objects including iron, copper, and bronze analogous to waste containers or the waste materials themselves; the degradation of glass and cementitious or bituminous material as an analogue of the wastes; the long-term evolution of physicochemical properties of cements and other building materials analogous to the structure of the disposal system; the decay and breakdown products of organic material and complexation with trace elements, analogous to waste degradation; and chemical interactions between buried objects and host rocks or soils that might be analogous to near-field processes. The study of these analogues provides several types of information that is useful to repository scientists. First, the analogues provide information about what types of materials are robust when subjected to the phenomena that a repository would be expected to be exposed to over very long time periods. This aids repository scientists in justifying selection of media within which to develop the geologic disposal facility, and understanding those natural and engineered barrier components which are likely to provide robust performance. Next, the analogues provide data about the processes themselves, particularly how they evolve through time, lead to changes (or lack thereof) in the materials, and how changes in the materials lead to concomitant changes in the manner by which the event and process phenomena affect the materials. This leads to yet a third way in which repository analogues are of use to repository scientists. The very long timescales that must be considered in order to assure safe disposal of used nuclear fuel and high-level radioactive waste in deep geologic disposal facilities raise issues about confidence in the projections of performance. Analogues provide data to help build confidence in the safety assessment models (e.g., supporting barrier capability discussions) and defend the long-timeframe performance projections that must be made with them to assess geologic disposal facility performance.

Safety assessments must consider long periods of time, large spatial scales, and complex, evolving conditions. Acceptance of the safety projections of a geologic disposal facility requires confidence that the models, computer codes, and databases used in the safety assessments are appropriate, reliable, and sufficiently realistic to provide an adequate and credible representation of the system. The bases for the conceptual models and databases used in safety assessments are the laboratory and field research programs, and their interpretations, which are, of necessity, carried out over relatively short periods of time. Understanding the past record of the natural system allows projection to future conditions. Natural and anthropogenic analogues provide the additional information to build confidence in extrapolating the estimated safety and performance of a disposal system over much larger spatial and temporal scales than that represented by the scientific investigations carried out by repository scientists. Analogue studies can shed light on both near-field processes and radionuclide migration in the far field and in the biosphere (International Atomic Energy Agency 1999b).

The initial and boundary conditions and other important parameters that lead to the formation of natural analogues are often not known precisely, which can limit the degree of accuracy with which the analogue can be interpreted. Analogue information is primarily qualitative because it is not possible, in most cases to quantify all relevant parameters in natural systems. This is an inherent limitation in the study of all complex natural systems where long-term processes have been active. For this reason, an analogue cannot be used, in general, in a quantitative sense for direct validation of a mathematical model such as a for radionuclide transport. Both qualitative and quantitative analogue information are used to test the robustness of a model and enhance confidence in its predictions (International Atomic Energy Agency 1999b).

Table 2-1 illustrates the main physical processes that may occur in a geologic disposal system grouped by the system component that would be affected. The processes encompass such phenomena as alteration, radiation effects, colloid and gas generation, dissolution, mineral transformations, diffusion, sorption, degradation, advection, and microbial activity. Many of these processes are coupled, that is, the ultimate action depends on relationships between two or more of the processes. Not all of these processes would be associated a given geologic disposal facility; for example, dissolution is more likely to be associated with salt than granite. Qualitative evidence showing that the materials and media selected for geologic disposal facility development are robust and long lived includes the information from analogue studies.

The evidence also shows that when these materials are subjected to the processes that a geologic disposal facility would be expected to experience, they tend to be stable and robust. Furthermore, the study of analogues provides data that allows the improved understanding of the conceptual foundation of the numerical models used in the safety assessments, providing confidence in the long-term safety projections. This is the basis for the conclusion that analogue evidence supports the rationale for the robustness of geological disposal as a waste management option. Detailed descriptions of example natural and anthropogenic analogues are presented in Appendix B.

Table 2-1. Main Processes That May Occur in a Geologic Disposal System Grouped by the System Component That Would Be Affected

Component	Process / Phenomena
Used Nuclear Fuel / High-Level Radioactive Waste	Alteration / dissolution
	Criticality
	Radiolysis (radiation effects)
	Speciation – solubility
	Colloid generation
	Gas generation
Waste Package	Corrosion
	Colloid generation
	Gas generation
Backfill and Sealing Materials Bentonite Buffer Cement	Dissolution – precipitation of impurities
	Dissolution – precipitation processes in variable temperature field: cementation
	Smectite – illite transformation
	Speciation – solubility
	Molecular diffusion in the bentonite barrier
	Sorption – adsorption and ion exchange
	Colloid generation and transport
	Gas generation and transport
	Cement degradation and generation and evolution of hyperalkaline plume
Geosphere	Advection and dispersion
	Fluid flow
	Groundwater – rock matrix interaction
	Speciation – solubility
	Redox state – Redox front
	Diffusion in rock matrix
	Molecular diffusion in clay formations
	Sorption – adsorption and ion exchange
	Precipitation – co-precipitation / dissolution
	Colloid generation and transport
	Gas generation and transport
	Microbial processes
	Coupled processes

Source: Ruiz Lopez et al. 2004.

2.1.2.5 The Value of a Staged or Cautious Facility Implementation Process

Development of a geologic disposal facility is an inherently cautious process, contributing to the robustness of geological disposal as a waste management option. A stepwise repository implementation process ensures that safety is re-evaluated at major stages in the development of a repository. In practice, each of the steps would include significant opportunity for affected parties to participate in a formal, public hearing process. For example, because the construction of a geologic disposal facility is a major federal action, an Environmental Impact Statement will be required, providing an opportunity for stakeholder participation and input in the evaluation of the potential for a site to perform safely.

The application of the siting guidelines during the site screening is cautious; the intention is to select a site that has a high likelihood of performing well if a repository is developed at a site under consideration. Because detailed site information is not likely to be available, sites should be assessed to limit the likelihood of encountering features, events, and processes that could lead to unacceptable consequences. Based on current generic knowledge, it appears that it should be possible to avoid or accommodate very unlikely disruptive natural processes and events and significant natural perturbations.

Past experience indicates that the site screening phase is an early point to evaluate the potential for a site to perform safely; "cautious, but reasonable" assumptions are appropriate, especially in the absence of detailed site specific data. The U.S. Nuclear Regulatory Commission is moving toward the risk-informed, probability-based approach of its current high-activity waste regulations and for other regulations (Vietti-Cook 1997).

Within each of the phases of the repository development process, scientific work continues to expand the understanding of a site. Periodic safety reevaluations will refine the basis for confidence that safety is being and will continue to be assured. This work is in addition to onsite inspections and continuing technical investigations by the regulator. Even after final repository closure there likely will be continued monitoring for a period of time to assure that conditions remain as expected and to provide physical security for the site.

Repository development will follow the applicable standards of engineering practice in design, review, and management. Standard practices typically involve pre-conceptual design activities, conceptual design, preliminary design, and detailed/final design, with appropriate documentation and discipline reviews at each step. Requirements will be implemented at prescribed steps in the process, particularly reviews of environmental impact and nuclear safety, to assure stakeholder acceptance and regulatory compliance. Quality assurance principles will be applied as is typical in the nuclear industry.

A phased approach to repository development has been recommended (National Academy of Sciences 2003) that would use pilot-scale prototype activities to verify technical analyses, design solutions, and project cost estimates before full-scale implementation. A similar approach has been recognized for repository closure, whereby the repository is closed in steps, with continuous collection of environmental data to evaluate waste isolation performance (for example, Andra 2005b). Commitment to very long-term monitoring has been recognized as an important driver for stakeholder acceptance, beginning early in the process with siting and characterization.

This stepwise implementation process allows there to be a reasonable approach to the basis supporting a given stage. Information must be developed during screening and characterization that is sufficient for construction authorization. The knowledge base will expand as scientific observation and testing continue during construction. During emplacement, as construction also continues, understanding will be further enhanced. If at any time during these stages there is evidence showing the basis for safety is not what it was thought to be, plans for retrieval and alternate storage of the radioactive wastes will be developed. The capability for retrievability will be maintained until final closure.

Deep boreholes have some disadvantages in terms of the difficulty and cost of retrieving waste after a borehole is sealed. Deep borehole emplacement options should be investigated to determine cost and technical potential for waste retrieval; if an area is proven adequate for deep borehole disposal in performance confirmation testing, closure of the hole would be allowable, and possibly not preclude postclosure retrieval.

Finally, the stepwise and cautious approach being taken allows modifications to be made over time, as warranted. Modifications could include ameliorating a heretofore insufficiently understood process or feature, perhaps, but more likely it will be the type of modification suggested by advances in technology and materials. Such advances may enhance safety or allow for safety to be unaffected while reducing the cost of the repository or allowing operations to be safely accelerated.

2.1.2.6 Summary: A Multi-Element Qualitative Rationale for Long-Term Safety

The rationale for the long-term safety and robustness of mined disposal systems in salt, clay, and granite, and deep borehole disposal in crystalline rock, has multiple elements. Historically, salt, clay, and granite geologic media have been chosen because of their potential to provide long-term safe protection of humans and their environment, and to do so in a robust manner. Their engineered and natural system components and their functions are understood well enough at this time to support a decision to move to regional, area, and site screening. Deep borehole disposal systems in crystalline rock appear equally capable of isolating wastes from the environment, however further research and development is needed to fully evaluate the emplacement and retrieval concepts. Based on current generic knowledge, it appears possible to avoid or accommodate disruptive natural processes and events.

The deep underground location of the geologic disposal facility along with siting away from any recoverable resources is less likely to attract human intrusion than a facility located closer to the surface or near recoverable resources; it will also reduce risks associated with terrorist access to or attacks on the waste. Long-term stability is assured by the geologic history of the types of formations considered, which have been in place for many millions of years. Potential host rock characteristics exist that can provide for relatively benign chemical environments that protect waste packages and other features of the engineered system, as well as the waste itself, from rapid corrosion and dissolution. Such a host rock formation provides a chemical environment that assures that a wide range of radionuclides will either not dissolve or will sorb onto the walls of fractures or capillaries.

Analogue evidence exists to support the selection of geologic disposal as a waste management option. Selected media and materials for geologic disposal facility components can be shown, through natural and anthropogenic analogues, to be robust, long lived and stable. These analogues provide convincing evidence that geologic disposal is an appropriate waste management option.

Development of a geologic disposal facility is an inherently cautious process. At each step along the way, there are opportunities for public involvement and participation in decision making, and external oversight and review. These opportunities are facilitated by this Generic Safety Case, which is intended to document issues of concern and how those issues are being resolved.

In total, this supports a robust total system safety expectation: uncertainty in the detailed performance of some features or processes is compensated for by the existence of the other features or processes.

2.2 Siting and Design Strategy

2.2.1 Introduction

An important aspect of the safety case approach is the concept of relying on multiple barriers. In identifying the elements of a safety case, Institute de Radioprotection et de Sureté Nucléaire (IRSN) (Bailey et al. 2011) notes that a statement of confidence in the safety of the disposal facility should describe how the multi-barrier / multi-function concept is effective and robust. A barrier can be defined as any material, structure, or feature that prevents, limits, reduces, or delays the rate of movement of water or radionuclides from the repository to the accessible environment, or prevents the release or substantially reduces the release rate of radionuclides from the waste. The use of both natural and engineered barriers constitutes a prudent approach to multiple barrier repository design. The combined geologic and engineered features that make up the postclosure repository system have several attributes that are important to keeping radionuclides away from humans. Because water is the primary medium by which radionuclides could be released from the repository, the barrier functions of the system primarily relate to the ability of the site and the design to limit or delay the movement of water and radionuclides.

The multiple barrier concept has received attention in the international radioactive waste disposal community. PAMINA (Bailey et al. 2011) states:

> *The safety case for a geological radioactive waste repository will be based on an understanding of the evolution and performance of the engineered barriers that contain the waste and the geological environment that isolates the waste from the human environment.*

The multiple barrier concept contributes to confidence in the safety assessment and safety case. Multiple barriers offer defense in depth which is achieved by a diversity of features and processes that act collectively, and often, independently. FANC/Bel V (Bailey et al. 2011) considers that "it is not the number and redundancy of the barriers as such that take on the greatest importance in terms of safety, but the fact of being able to depend on different mechanisms and/or components to provide safety functions."

The contributions of each barrier can be organized by identifying its safety functions (Organisation for Economic Co-operation and Development 2007). PAMINA (Bailey et al. 2011) defines safety functions as those properties of the engineered and natural barriers that provide safety. The *International Atomic Energy Agency: Safety Standards for Geological Disposal of Radioactive Waste* (International Atomic Energy Agency 2006) describes requirements for multiple safety functions:

> *The natural and engineered barriers shall be selected and designed so as to ensure that post-closure safety is provided by means of multiple safety functions. That is, safety shall be provided by means of multiple barriers whose performance is achieved by diverse physical and chemical processes. The overall performance of the geological disposal system shall not be unduly dependent on a single barrier or function.*

While different terminology is used in different countries, safety functions fall into some general categories (Bailey et al. 2011):

- **Stability/Isolation Safety Function**—Two subgroupings are identified:
  - Isolating the waste from non-anthropogenic future events and climate changes, and which thus contributes to the stability of the repositories' near-field conditions and to the longevity of the natural barriers. The deep borehole disposal concept is particularly robust with respect to this safety function.
  - Reducing the probability of and consequences from anthropogenic events such as future human actions that might result in inadvertent intrusions into the sealed repository.
- **Containment—**Preventing or limiting groundwater from coming into contact with the waste. In the case of disposal in crystalline rock or clay/shale formations this safety function is provided by the engineered barrier system. In the case of disposal in salt formations much of the containment function may be provided by the natural barrier system.
- **Limited and/or Delayed Releases—**These safety functions begin to dominate once the containment functions deteriorate, for example, when waste packages are breached as a result of corrosion. This is a major function of the natural barrier system as well as components of the engineered barrier system and provides for the long-term barrier capability of geologic disposal.

The concept of “important to waste isolation” may be used to define features that contribute to barrier safety functions. A barrier’s feature can be important to waste isolation if it meets two conditions (Sandia National Laboratories 2008b): (1) the feature is associated with one or more processes or characteristics classified as important to barrier capability; and (2) the feature is a significant contributor to the barrier capability relative to the other features of the barrier. In addition, a feature may be classified as important to waste isolation if it is one of the engineered features of the geologic repository whose function is to prevent or mitigate the consequences of potential disruptive events.

The demonstration of barrier capability and evaluation of safety functions consider both qualitative and quantitative information. Qualitative information includes a summary of the events and processes acting on each barrier feature that contribute to barrier capability. Site characterization, field tests and laboratory experiments also provide quantitative understanding that barriers work together to perform their postclosure functions. Quantitative information includes subsystem analyses using the safety assessment model or process level models focused on specific barriers or barrier feature and processes. Measures for quantifying the results of safety assessment calculations, mainly dose and risk, have always been used. There is a wide international consensus today that it is necessary to use complementary safety indicators to improve the understanding of the system and to support the safety case. The long-term repository safety should be assessed using several independent indicators. These indicators may be used to quantify or demonstrate the performance of subsystems or single barriers, or of the total system, in order to build confidence in system or component performance over long time periods.

In the U.S., evaluations of barrier performance were made using the safety assessment model and analysis of system and subsystem performance under various imposed conditions, including failure of some of the engineered components and consideration of disruptive events (U.S. Department of Energy 2008b). The evaluations examined the performance of features as well as the contribution of processes acting on and within these features, and the impact of events on the features. The features and processes were evaluated with respect to how they (1) prevent or substantially reduce the rate and amount of water that may seep into the repository drifts and, in turn, reduce the quantity of water potentially contacting the waste form (containment), and (2) prevent or substantially reduce the release rate of radionuclides from the waste and prevent or substantially reduce the rate of movement of radionuclides from the repository to the accessible environment (limited and/or delayed releases). Safety confidence is enhanced by designing multiple safety capabilities or functions into the system. The engineered barriers must be designed such that they work synergistically, or in combination, with the natural barriers in which the repository is sited.

The general safety functions provided by engineered barrier components or “features” are also common to the natural barrier features or components, and include the three primary functions discussed above (Bailey et al. 2011): isolation, containment, and limited and/or delayed releases. Many of the physical and chemical processes at work in the engineered barriers, which can serve to both help or hinder the operation of the safety functions or barrier capability, are also common to the natural barrier system components, and include those processes related to water movement and associated degradation or enhancement of safety functions and/or physical components. In particular, the movement of water in the repository can either hinder or promote safety functions. For example, corrosion of waste containers by water intrusion degrades the containment safety function of the container, whereas geochemical immobilization (e.g., retardation, precipitation) of radionuclide elements in the engineered components (after container degradation) is a safety function that is promoted by the presence of water. These are called complementary safety functions (Organisation for Economic Co-operation and Development 2004).

2.2.2 Siting Strategy

2.2.2.1 Natural Barrier Considerations

The natural barrier system is a set of durable barriers of the disposal system, as described in Section 2.1.2.1 and indicated in Figure 2-1. In the absence of external events, the properties of the natural barriers are not expected to change in any significant way, even during long timeframes. The natural barrier system contributes to all three of the safety functions: isolation, containment, and limited and/or delayed releases.

In the event that radionuclides are released from the engineered barriers, flowing groundwater in the natural system transports the radionuclides either in solution (dissolved) or in suspension, bound to very small particles known as colloids. Colloids may be small enough to travel with flowing water through fractures and through pores in the unfractured matrix portion of the natural system. The processes relevant to the transport of radionuclides in the natural system include advection, matrix diffusion, dispersion, colloid-facilitated transport, sorption, and radioactive decay and ingrowth.

2.2.2.1.1 Features of the Generic Natural System

While many features of the natural barrier system are site specific, generic categories of features can be identified. The following identifies a number of natural barrier system features along with parameters that influence the barrier capability of the natural system in general. More attention is provided to the hydrologic features and parameters, because many of the other types of relevant natural system features (topographic, lithologic, morphologic, and stratigraphic) have hydrologic consequences.

- **Topographic Features**—Associated with the surface such as slope, presences of washes and outcrops, soil depth, vegetation covering. These features influence the infiltration rate, which affects the rate and amount of water that may contact the disposal system.

- **Lithologic Features—**Related to rock type, mineralogy, and composition. These features influence the geochemistry in the natural system, which contribute to the chemistry of the water that may contact engineered components as well as to the transport of radionuclides away from the engineered barrier system.

- **Morphologic Features**—Associated with the characteristics, configuration, and evolution of rocks and land forms. Geomorphically relevant processes include weathering and erosion process such as chemical dissolution, mass wasting, glacial action, tectonism, and volcanism. These features contribute to the function of stability and the ability of the natural system to respond to external natural events.

- **Stratigraphic and Structural Features**—Associated with rock layers. A common goal of stratigraphic studies is the subdivision of a sequence of rock strata into mappable units, determining the formation and alteration time relationships that are involved and correlating units of the sequence with rock strata elsewhere. Stratigraphic, structural, and associated hydrologic properties have significant effects on natural barrier system flow and transport processes due to (1) the contribution of faults in conducting flow or acting as a barrier to flow; (2) the effects of folding of layers on flow paths; and (3) the different flow characteristics of adjacent layers. For example, a lower effective conductivity of the surface bedrock will tend to increase water storage in the surficial soil and increase the effectiveness of runoff and evapotranspiration, thereby reducing the rate of net infiltration into the subsurface.

  Adjacent strata or regions of rock that have dramatically different hydrologic properties can influence natural system barrier performance. For example, if flow is going from a more permeable region to one of lesser permeability, a portion of the flow may move perpendicular to its original direction. In unsaturated rock, if flow is from a region where fracture flow dominates to one where matrix flow dominates, then flow may be attenuated or dampened by virtue of the larger storage capacity of the matrix. In unsaturated rock rock-property contrasts between sublayers or regions may produce capillary barriers at the interface that promote change in flow direction (Montazer and Wilson 1984).

- **Rock Properties**—Have a significant effect on the rate of radionuclide movement through their influence on the transport properties (notably, porosity, permeability, sorption, and for fracture dominated flow such as could occur in granites, the flowing interval spacing, matrix diffusion coefficient, and fracture porosity). Flowing interval spacing represents the distance between fractures, or sets of fractures, that transmit significant quantities of groundwater. The thermal properties (notably thermal conductivity) influence the rate of dissipation of decay heat from the waste.
- **Hydrologic Features**—Associated with the flow of water through the rock. Some of these features include the degree of saturation, the degree of fracturing, the degree of consolidation, and the type of formation.
  - *Degree of Fracturing*—Fracture characteristics are important to the barrier capability of the natural barrier system because, if fractures are present and well-connected over large distances (hundreds of meters to kilometers), groundwater flow often occurs primarily within the fracture network. Unfractured media are generally more favorable to the barrier capability of the natural barrier system than when fractures are present. Thus fracture networks control the movement of dissolved and colloidal radionuclides. The rate of flow and the extent of transport in fractures are influenced by such characteristics as orientation, aperture, asperity, spacing, fracture length, connectivity, porosity, and the nature of any linings or infill. Open fractures in the rock will tend to increase the rocks effective hydraulic conductivity and result in an increased rate of water movement (infiltration into the disposal system and, possibly advective radionuclide transport away from the disposal system). Fractures at or near the surface may be partially or completely filled with minerals or other material, which could substantially reduce infiltration.

    Faults may provide fast flow and radionuclide transport pathways through the natural barrier system by affecting groundwater flow paths, influence the anisotropy in permeability, and can enhance dispersion by increasing permeability heterogeneities. Depending on location and orientation, faults may also act as barriers to flow. Faults may contain highly fractured rock and have high permeability or may contain fine-grained fault gouge and have low permeability. If they are filled with fine grained material, they may have a lower permeability than the surrounding intact rock. This may result in more flow occurring through rock matrix compared to the fracture. The content of faults, fractures, and the matrix are affected by the precipitation of minerals in the pores and fractures of the host rock. This reduces formation and fracture connectivity, porosity, permeability, and may provide for radionuclide sorption.
  - *Degree of Consolidation*—Unconsolidated sediments have little or no mineral cement or matrix binding its grains. They are materials produced by weathering, sediment deposition, biological accumulation, and/or human and igneous activity. They are often associated with surficial materials but may occur below the surface as well. They can vary from clay to sand to gravel and have connected pore spaces that allow groundwater to be stored and transported. Examples include alluvium, glacial deposits, and some ocean deposits. Unconsolidated sediments may provide considerable capacity to retard the migration of radionuclides. Geologic processes, including compaction due to increasing depth of burial, or decrease in pore volume due to movement of water and deposition of minerals, can increase the degree of consolidation. Consolidated sediments are materials that have been cemented together. Examples include sandstone and shale.

- *Type of Formation*—Rock formations may vary in their ability to store and transmit water. An aquifer is a water saturated formation that can readily store and transmit water because it has well developed and interconnected porosity and is permeable. Precipitation or infiltration from surface water bodies adds water (recharge) into the porous rock of the aquifer. Aquifers can be quite extensive (tens to hundreds of miles) and can feed many wells and streams. Aquifers may be isotropic or anisotropic. In isotropic aquifers or aquifer layers the hydraulic conductivity is equal for flow in all directions, while in anisotropic conditions it differs, notably in horizontal and vertical directions. There are two end members in the spectrum of types of aquifers: confined or unconfined. A confined aquifer is an aquifer below the land surface that is saturated with water and has layers of less permeable material both above and below. Water in the pores of a confined aquifer may be under sufficient pressure so that when the aquifer is penetrated by a well the water will rise above the top of the formation. An unconfined aquifer is an aquifer whose upper water surface (water table) is at atmospheric pressure, and thus is able to rise and fall. Water-table aquifers are usually closer to the earth's surface than confined aquifers are. The value of specific yield obtained from an aquifer test can be used to determine if an aquifer is confined or not. Confined aquifers have very low storativity values (less than 1% of bulk volume) which means that the aquifer is storing water using the mechanisms of aquifer matrix expansion and the compressibility of water, which typically are both quite small quantities. Unconfined aquifers have storativities greater than 1% of bulk volume and they release water from storage by the mechanism of actually draining the pores of the aquifer, releasing relatively large amounts of water.

  An aquitard is a formation that restricts the flow of water one aquifer to another. Aquitards are composed of layers of either clay or non-porous rock with low hydraulic conductivity. An aquitard can sometimes, if it has very low permeability, be called an aquiclude. Semi-confined aquifers with one or more aquitards work as an anisotropic system, even when the separate layers are isotropic, because the effective vertical and horizontal hydraulic conductivities differ.

- *Degree of Saturation*—The degree of saturation (fraction of pore space occupied by water) influences the barrier capability of the natural barrier system. A formation can be can be divided into two regions: the saturated (phreatic) zone (e.g., aquifers, aquitards), where all available spaces are filled with water, and the unsaturated (vadose) zone, where gas and water share the pore space but can be filled with more water. Water that is in direct contact vertically with the atmosphere through open spaces in permeable material is called unconfined water. Confined water is separated from the atmosphere by impermeable material and the pressure of the water can be greater than if it were unconfined.

  Unsaturated conditions occur above the water table where the pressure head is negative (absolute pressure can never be negative, but gauge pressure may be) and the water that incompletely fills the pores of the aquifer material is under suction. Under unsaturated conditions water is held in place by surface tension forces that produce a capillary pressure difference across the water gas interface. The strength of this capillary force depends on soil pore size and other parameters.

  Unsaturated media generally result in lower infiltration and water flow away from the repository. In unsaturated media the presence of gas in the pore space interrupts the continuity of the liquid phase and the capillarity and surface tension at the interface of the gas and liquid phases decreases the ability of liquids to flow. If the unsaturated media is not isolated from the surface, the geochemistry and radionuclide chemistry are generally in an oxidative environment, which may result in conditions more conducive for radionuclide transport. In deep saturated rock, reducing chemistry often dominates, which increases the barrier capabilities of the natural barrier system for many potentially important radionuclides because their solubility limits are lower and retardation is greater.

These features, and the processes that act upon them, affect the behavior of the natural system. Relevant processes in the natural system, discussed further in Section 3.3, include flow in the surface and subsurface environments, geochemical processes, and radionuclide transport by means of advection, diffusion, dispersion, sorption, and colloid transport.

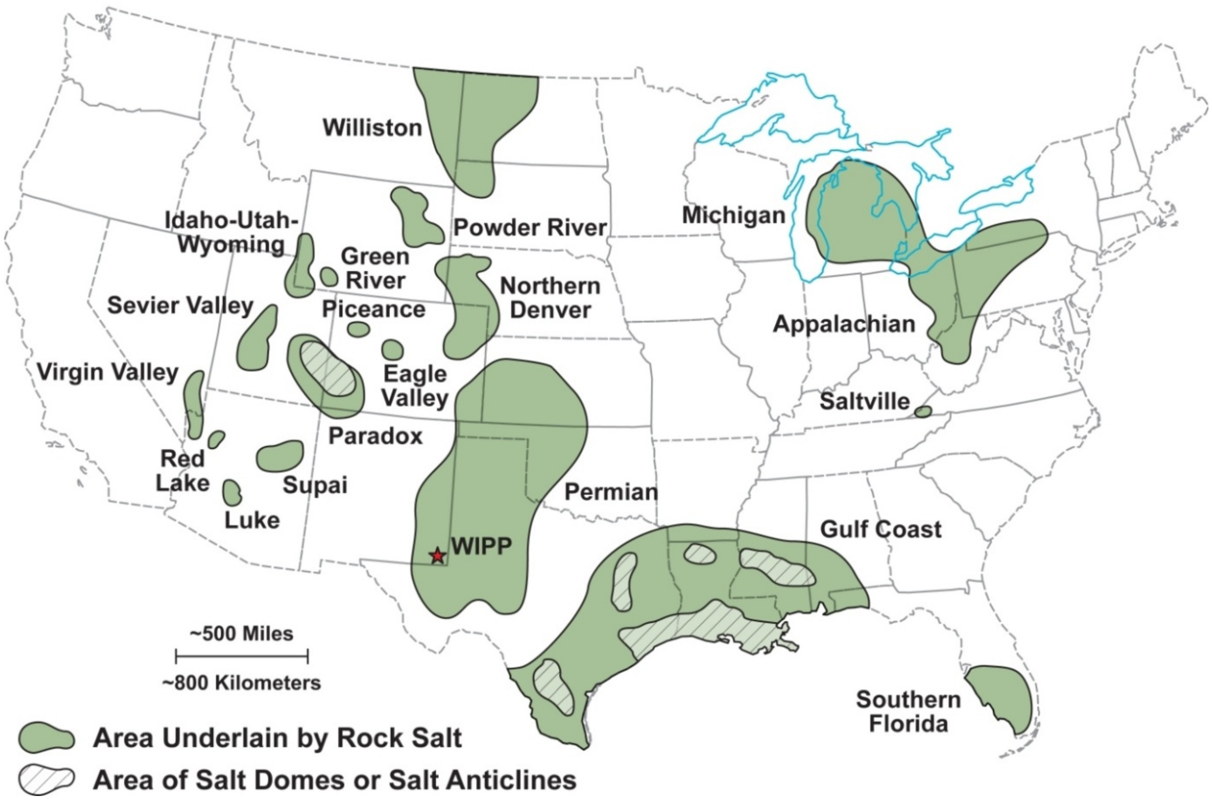
2.2.2.2 Available Geologic Media for Siting

A number of preliminary studies and papers addressing the available area in the U.S. for siting a geologic disposal facility have been completed. The most recent of these studies, performed as part of the Used Fuel Disposition Campaign, have been generic in nature and have not focused on specific sites; they have focused on identifying potential areas that have a higher likelihood of serving as appropriate locations for a deep geologic repository. There is no intention of suggesting that other acceptable sites could not also be identified. The studies have focused on some general attributes of the geologic setting that would be desirable for siting. Formations with low hydraulic conductivity and high thermal conductivity may perform well. In addition, a medium such as salt or a plastic clay which is capable of self-sealing, whereby excavations would close naturally, was seen to be desirable. A low-permeability medium would be more likely to lead to diffusion-dominated transport; also, chemically-reducing conditions in the host rock are seen as desirable (Hardin et al. 2011b). The Used Fuel Disposition Campaign studies are evaluating the availability of salt, clay, and granite media for development of a mined repository and crystalline rock for a deep borehole disposal facility.

These four disposal options are discussed in the following subsections. The discussions include subjective comparative assessments about a number of parameters (e.g., thermal conductivity, permeability, strength) that may be relevant to the performance of a geologic disposal facility. The descriptions are comparative between the different geologic media. For example, a statement that something is high indicates that it is at the end of the range, while medium means it is in the middle of the range of the parameter values for the three media. The information is brought together for comparative purposes in a Section 2.2.2.2.5.

2.2.2.2.1 Salt

The use of salt formations for nuclear waste disposal has been widely pursued for more than 50 years; in the U.S. it is traceable to the recommendations of the National Research Council in 1957 to use a salt mine for disposal of the high-level radioactive waste from reprocessing (National Academy of Sciences 1957). Screening of the entire U.S. in the 1960s and 1970s identified large regions underlain by rock salt of sufficient depth and thickness to accommodate a repository (Hansen et al. 2010). Figure 2-4 is a map of salt deposits showing both bedded and domal salt. The domal salt is shown as cross-hatched areas. As shown in Figure 2-4, the conterminous U.S. has many thick and/or laterally extensive salt formations, including bedded and domal salt. Four major regions of the U.S. where salt formations are found include (1) the Gulf Coast, (2) the Permian Basin, (3) the Michigan-Appalachian Region, and (4) the Williston Basin. Domal salts are found in the Gulf Coast region and Paradox Basin, and bedded salts are generally present in the remaining three major salt regions of North America. The salt basins tend to look geographically similar to the shales because they are in fact formed in similar depositional environments (Hardin et al. 2011b).



Source: Hardin et al. 2011b.

Figure 2-4. Map of Salt Deposits in the United States

Site screening efforts by the U.S. Department of Energy in the 1980s recognized the following regions:

- **Salt Domes in the Gulf Coast**—The primary initial screening factors used to identify potentially favorable locations were the depth to the top of the dome and present use for gas storage or hydrocarbon production. Siting guidelines and the related evaluation reduced the list of over 500 salt domes to seven potential repository locations, with further screening resulting in the identification of the Cypress Creek, Richton, and Vacherie domes (Figure A-1) as potentially acceptable sites.
- **Bedded Salt in Utah**—The primary initial screening factors used to identify potentially favorable locations were the depth to the salt, the thickness of the salt, proximity to faults and boreholes, and proximity to the boundaries of the dedicated lands. The thickness of the salt, the thickness of the layers above and below the depth of a repository, and the minimum distance to salt-dissolution features were considered the most critical geologic discriminators. Using the siting guidelines, Davis Canyon and Lavender Canyon (Figure A-1) were identified as potentially acceptable sites.
- **Bedded Salt in West Texas and Southeastern New Mexico**—The Permian bedded-salt deposits in the Texas panhandle, eastern New Mexico, and western Oklahoma had been identified as potentially suitable for waste disposal. The primary screening factors were the depth to and the thickness of the salt, faults, seismic activity, salt dissolution, preexisting boreholes, underground mines, proximity to aquifers, mineral resources, and conflicting land uses, such as historical sites and state or national parks. All the evaluated subbasins contain salt beds of adequate thickness and depth. Using the siting guidelines, locations in northeastern Deaf Smith and north-central Swisher counties in Texas (Figure A-1) were identified as potentially acceptable sites.

The U.S. has supported significant investigations in salt in the past, including Project Salt Vault, the Carey mine in Western Kansas, Avery Island, Louisiana, and the WIPP, where some limited thermal testing has been done. Disposal of nuclear waste in salt remains a viable concept in the U.S., as has been demonstrated by virtue of more than 11 years of successful operations at the WIPP near Carlsbad in New Mexico (U.S. Department of Energy 2011a). Currently, there is also interest in a salt disposal option in Germany (Appendix C, Section C-3.1.3).

The thermal conductivity of salt is high relative to other potential host media, which generally would be favorable for disposal because conduction of heat away from the wastes will limit temperature build-up. The resistance to heat is also high, indicating that there is little potential for the heat of the waste to damage the salt rock; together, these should generally allow for higher thermal loadings. The potential for salt to be dissolved in the presence of water is high. Repository disposal facilities may be located deep beneath the ground surface, ensuring reducing geochemical conditions surrounding the repository rocks that would be expected to limit the solubilities of radionuclides in the waste forms. However, the sorptive properties of salt, which slow or limit the migration of any radionuclides that might escape from a degraded waste package or other engineered barriers are low.

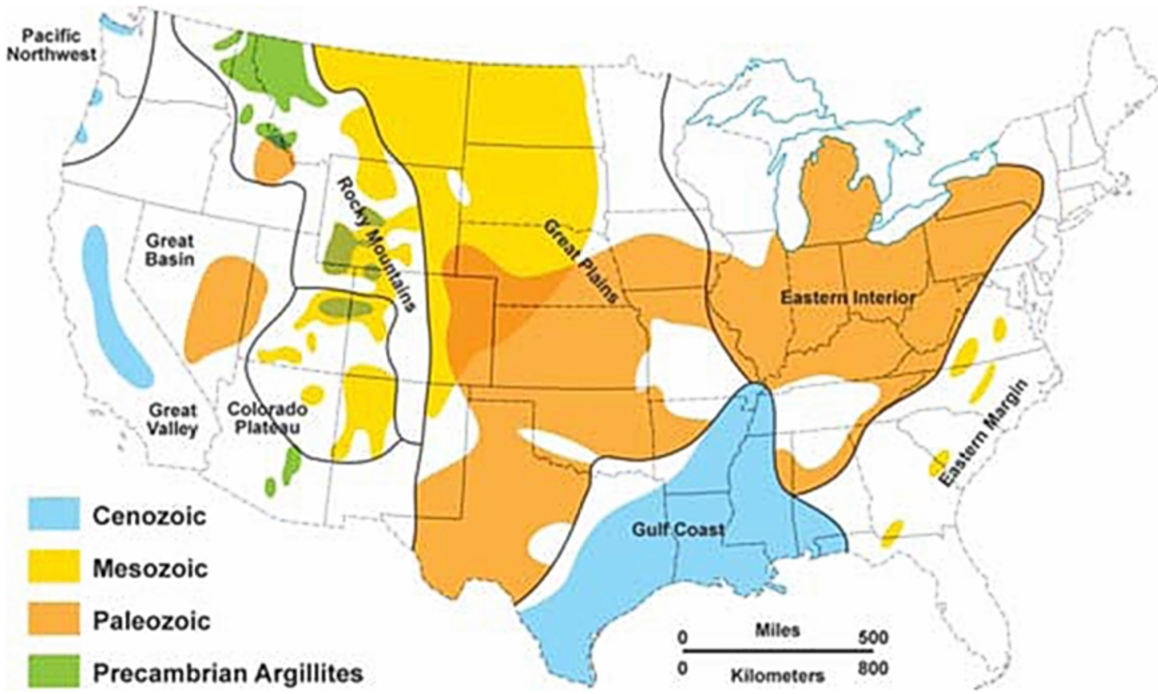
There is significant experience in mining in salt. The in-situ stress state is likely to be isotropic, or, the same in all directions, which is a benefit in developing a facility layout. Salts are visco-plastic and time dependent deformation of underground openings can be expected; because of this deformation, artificial reinforcement may be required in order that underground openings can be constructed and maintained safely during repository operations period (U.S. Department of Energy 2004b).

Salt rock masses are expected to be highly impermeable, which would ensure that the natural barriers would contribute significantly to containment of the radionuclides by preventing movement of any radionuclides that could escape from the waste packages and other engineered barriers. Such low permeability would also prevent water that from contacting the wastes. Fractures in salt are likely to be self-healing due to creep and the deformable nature of salt.

2.2.2.2.2 Clay/Shale

Clay/shale formations in the U.S. meeting the general siting considerations for depth, thickness, and other criteria summarized in Section 2.3.2 are common in the U.S., as shown in Figure 2-5. There are potentially significant differences in rock characteristics included in this category of sedimentary rock, as discussed in a recent study of the performance of clay/shale repositories for high-activity waste in the U.S. (Hansen et al. 2010). Gonzales and Johnson (1984) concluded that the most desirable host rocks should be between 300 and 900 m below ground level, at least 75-m thick, relatively homogeneous in composition, and in an area of low seismicity and favorable hydrology that is not likely to be intensively exploited for subsurface resources. High clay content is preferred to ensure low permeability and plasticity.

Figure 2-5 also indicates the geologic age of the shale provinces shown, including Cenozoic, Mesozoic, Paleozoic, and pre-Cambrian. Generally, the induration or the lithic character of the clay or shale increases with its age and depth of burial; the softest and most plastic clays or shales would be expected to be the youngest, those are shown in blue. There are other large shale basins available in the continental U.S. The U.S. also had an active shale repository research and development program in the 1970s and 1980s, which included some laboratory and in-situ scale thermo-mechanical tests. Here again, confidence in the ability to eventually develop a geologic disposal facility in clay/shale is strong because of progress in other countries, particularly the French program (Appendix C, Section C-3.2.3).



Source: Hardin et al. 2011b.

Figure 2-5. Shale Formations in the United States

Intact or unfractured clay or shale masses are expected to have low to very low permeability, which would ensure that the natural barriers would contribute significantly to isolation of the radionuclides by resulting in slow movement of any radionuclides that could escape from the waste packages and other engineered barriers. Low permeability would also limit the amount of water that could contact the wastes. Fractures in clay or shale are likely to be self-healing in the presence of water.

The thermal conductivity of clay/shales is low, generally resulting in a need to spread the waste over larger areas. The resistance to heat is also low, indicating that there is a potential for the heat of the waste to damage the rock, resulting generally in a need for lower thermal loadings. Figure 2-5 indicates that there are likely to be sufficient large clay/shale formations available to accommodate large repository facilities. Furthermore, while the potential for shale to be dissolved in the presence of water is low, some shale will slake, or disintegrate in the presence of water. Testing to determine whether or not a particular shale will slake is straightforward. Repository facilities could be located deep beneath the ground surface in an environment of reducing chemical conditions; such geochemical conditions in the repository would limit the rate that the waste packages might corrode or the solubilities of the radionuclides released from the waste forms. Additionally, the strong sorptive properties of clay and shale would slow or limit the migration of any radionuclides that might escape from a degraded waste package or other engineered barriers. There may be minimal need for engineered barriers to provide additional confidence in the isolation capability of the repository.

Hansen et al. (2010) note that clay/shale rock masses have properties that could present some challenges to developing an engineered facility. The strength of clay/shale is generally low to medium and its geotechnical character leads to a need for artificial support to maintain stable openings in facilities in shale. The in-situ stress state could be oriented unfavorably with respect to existing fracture systems, which could be a consideration in developing a facility layout. Clay/shales range from plastic to brittle, and time-dependent deformation of underground openings can be expected along with an expectation that artificial reinforcement will be required in order that underground openings can be constructed and maintained safely for repository operations.

Characterization of possible clay/shale formations for high-activity waste repositories in the U.S. has not been undertaken. However, from the 1970s until the mid-1980s Oak Ridge National Laboratory led the U.S. research and development efforts for shale repository investigations. Oak Ridge National Laboratory directed testing programs specifically to characterize a few accessible shale formations, collecting repository-relevant physical, mechanical, mineralogical, and hydrological information. Testing efforts to characterize thermo-mechanical responses of selected shale formations were elementary. More lab and field work would be needed to characterize any particular clay/shale site. International communications and collaborations with France and Switzerland may be a key source of information (Hansen et al. 2010).

Clay/shale formations that could host a high-activity radioactive waste repository in the U.S. span a range of lithologies with different physical, mechanical, hydrological, chemical, and mineralogical properties. Two primary concerns for evaluating the effects of repository construction, operation, and long-term waste isolation performance are geomechanical response and fluid flow. The specific properties that control these important rock characteristics would be among those closely examined at the time of site selection and characterization (Hansen et al. 2010).

2.2.2.2.3 Granite

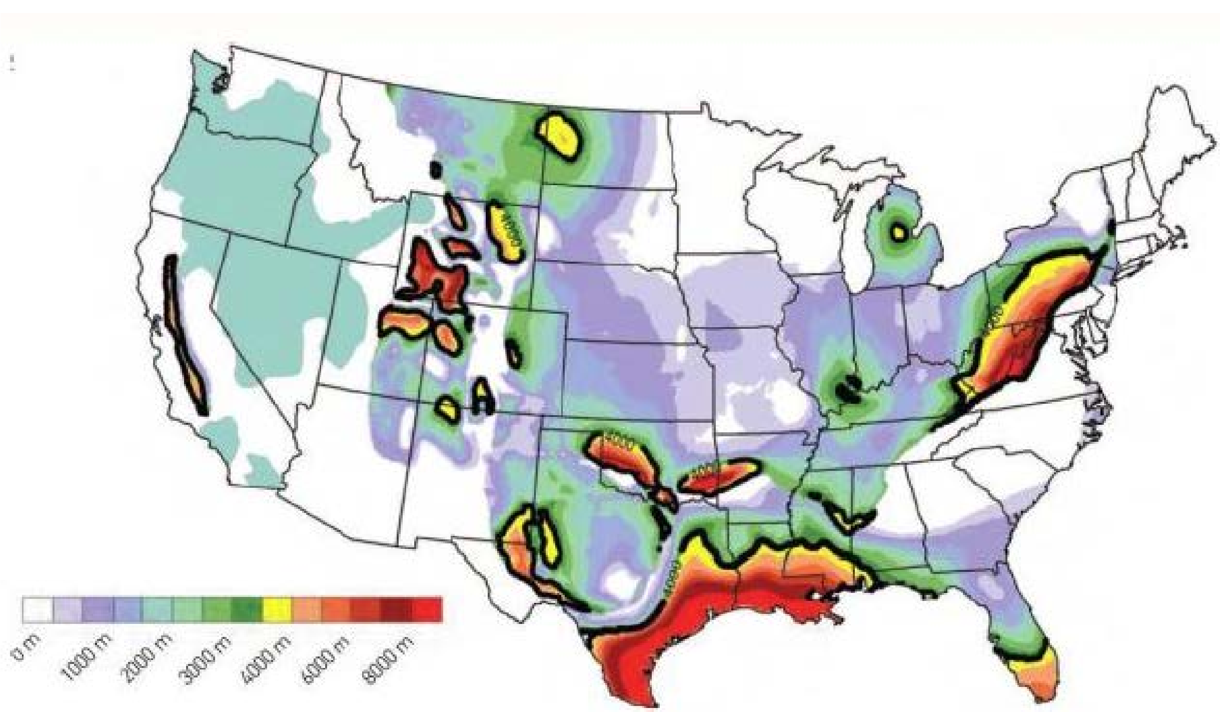
The 48 conterminous states have an abundance of granitic formations. Several countries have determined that granite formations are adequate for mined geologic disposal and Sweden and Finland are pursuing geologic disposal in granite. The U.S. had a research and siting program for crystalline rock until the late 1980’s. A look at granite going forward would probably include fractured and unfractured rock, and saturated and unsaturated conditions. Fractured and saturated conditions would probably lead to circumstances most resembling those that have been found in Scandinavia. There is reason to believe that a suitable site can be found in the U.S., especially considering progress in other countries. Figure 2-6 illustrates granite outcrops in the U.S. (Hardin et al. 2011b).

In addition to the granitic rocks outcropping at the surface, Figure 2-7 illustrates that a large portion of the central U.S. is underlain by granitic rocks as shallow as 500 to 1,000 m (Massachusetts Institute of Technology 2006), which is at depths likely to be suitable for a repository. These figures indicate that there likely is adequate granite available to develop an underground disposal facility.

With the availability of such large volumes of granitic rock, it should be possible to identify many areas of the country with granite formations possessing geological attributes that are potentially favorable to geologic disposal. Intact or unfractured granite masses are expected to have very low permeability, which would ensure that the natural barriers would contribute significantly to isolation of the radionuclides by resulting in slow movement of any radionuclides that could escape from the waste packages and other engineered barriers. Low permeability would also limit the amount of water that could contact the wastes. Fractured granite masses are likely to be more permeable (Hansen et al. 2011). Most repository designs in granite include a clay buffer material to eliminate or reduce advection and promote diffusive transport in the engineered barrier system.

Source: Hardin et al. 2011b.

Figure 2-6. Granite Outcrops in the United States



Source: Massachusetts Institute of Technology 2006.

Figure 2-7. Depth to Basement Crystalline Rock in the United States

The thermal conductivity of granites is medium, generally resulting in a need to spread the waste over larger areas. The resistance to heat, however, is high, indicating that the heat of the waste is not likely to alter the mineralogical, chemical, or physical characteristics of the granite rock. Furthermore, the potential for granite to be dissolved in the presence of water is very low. Repository facilities could be located deep beneath the ground surface, ensuring reducing chemical conditions; such geochemical conditions in the repository are typically favorable for limiting how the waste packages might corrode or the solubilities of the waste forms. Additionally, the sorptive properties of granite, which slow or limit the migration of any radionuclides that might escape from a degraded waste package or other engineered barriers are medium to high. Engineered barriers will provide additional confidence in the isolation capability of the repository.

Granite rock masses have properties that are conducive to developing an engineered facility. There is considerable experience in mining in granitic rocks. The high strength of unfractured granite and its geotechnical character lead to a high potential for developing stable openings in granites. With significant natural fracturing, however, the potential for developing stable, unsupported, openings is low. The orientation of the in-situ stress state will be a consideration in developing a facility layout; if oriented unfavorably, especially with respect to existing fracture systems, additional ground support could be required. Granites are strong and brittle, with little expectation for time dependent deformation of underground openings, and an expectation that underground openings can be constructed and maintained safely and economically for repository operations (Hansen et al. 2011).

2.2.2.2.4 Deep Borehole Disposal

Deep borehole disposal is a meaningful alternative for a geologic disposal system, in part due to the wide expanse of basement crystalline rock at design depth in the 48 conterminous states. Figure 2-7 shows a contour map depicting depth to crystalline basement. For deep borehole disposal there is a need to find the basement within a range of approximately 3 km of the surface, or less. Low permeability, high salinity, and geochemically-reducing conditions at many locations in the deep crystalline basement rock are expected to limit significant fluid flow and radionuclide transport. Crystalline rock bodies at great depth are expected to have very low permeability, which would ensure that the natural barriers would contribute significantly to isolation of the radionuclides by resulting in slow movement of any radionuclides that could escape from the waste packages and other engineered barriers. Low permeability would also limit the amount of water that could contact the wastes.

Though the relatively high temperatures and salinities of deep fluids by themselves could accelerate the corrosion of steel casing pipes, waste packages or canisters, fuel assemblies, and the waste itself, the scarcity of oxygen is expected to slow the oxidation of used fuel and corrosion processes. The geochemical behavior of the projected waste inventory in the deep borehole environment sets limits on the stability of the uranium in the used fuel matrix and on radionuclide transport to the biosphere. Nonetheless, the seal components of the engineered barrier system may play an important role in performance.

Given the low expected volumetric thermal loading for deep borehole disposal, the thermal properties of the deep crystalline rock are expected to ensure that there will not be thermal issues related to the disposal. The resistance to heat is high, indicating that the heat of the emplaced waste is not likely to alter the mineralogical, chemical, or physical characteristics of the crystalline rock (rock melt techniques for borehole drilling are not considered here). The figures indicate that there are sufficient large crystalline rock bodies available to accommodate the deep borehole facilities. Furthermore, potential for crystalline rock to be dissolved in the presence of water is very low. Borehole disposal is by definition located deep beneath the ground surface, ensuring reducing chemical conditions; such geochemical conditions surrounding the boreholes are typically favorable for limiting how the waste packages might corrode or the solubilities of the waste forms. Additionally, the sorptive properties of crystalline rock, which slow or limit the migration of any radionuclides that might escape from a degraded waste package or other engineered barriers are medium to high. Seal components of the engineered barrier system are expected to provide confidence in the isolation capability of the borehole disposal system.

Crystalline rock masses have properties that are conducive to developing a borehole disposal system. There is limited experience in drilling (large diameter especially) deep boreholes in crystalline rocks, although there is experience in drilling deep boreholes for oil and gas exploration and extraction in non-crystalline rock. The high strength of crystalline rock and its geotechnical character lead to a high potential for developing stable boreholes in deep crystalline rocks. The in-situ stress state is likely to be anisotropic, and could be a consideration in developing a deep borehole disposal facility due to borehole breakouts. Crystalline rocks are strong and brittle, with little expectation for time dependent deformation of the deep boreholes (Hansen et al. 2011).

2.2.2.2.5 Summary

The salient points of the evaluations of the four disposal options are summarized in Table 2-2. Not all of the properties listed in Table 2-2 are of equal concern; however this table is constructed for general considerations in generic repository evaluations. Details, characteristics, and attributes in Table 2-2 could be expanded extensively. However, other factors of the geologic environment and domain would be considered during site screening, and it must be remembered that preparation of a table such as this places focus on subsystem components. Risk-informed considerations are expected to promote more of a total system focus for safety assessments. Based on decades of international experience in repository development, it is highly probable that a suitable repository can be developed for any or all of the four options discussed above (Hansen et al. 2011; after BMWi 2008). The preceding discussion principally has been focused on geologic conditions for disposal; engineered systems may be applied that will contribute to containment of waste for a significant period of geologic time (hundreds of thousands of years to a million years of containment barrier contribution).

Table 2-2. Relative Attributes of Disposal Options

Property	Salt	Shale	Granite	Deep boreholes
Thermal conductivity	High	Low	Medium	Medium
Permeability	Practically impermeable	Very low to low	Very low (unfractured) to permeable (fractured)	Very low
Strength	Medium	Low to medium	High	High
Deformation behavior	Visco-plastic (creep)	Plastic to brittle	Brittle	Brittle
Stability of cavities	Self-supporting on decade scale	Artificial reinforcement required	High (unfractured) to low (highly fractured)	Medium at great depth
In situ stress	Isotropic	Anisotropic	Anisotropic	Anisotropic
Dissolution behavior	High	Very low	Very low	Very low
Sorption behavior	Very low	Very high	Medium to high	Medium to high
Chemical	Reducing	Reducing	Reducing	Reducing
Heat resistance	High	Low	High	High
Mining experience	High	Low	High	Low
Available geology*	Wide	Wide	Medium	Wide
Geologic stability	High	High	High	High
Engineered barriers	Minimal	Minimal	Needed	Minimal

Favorable property Average Unfavorable property

2.2.3 Design Strategy

2.2.3.1 Engineered Barrier Considerations

As discussed in Section 2.2.1, the design of a geologic repository should include multiple barriers, both natural and engineered. Geologic disposal of used nuclear fuel and high-level radioactive waste is predicated on the expectation that one or more aspects of the geologic setting will be capable of contributing to the isolation of radioactive waste and thus be a natural barrier important to waste isolation. While there is extensive information about the character of geologic formations, covering many millions of years, this record is subject to interpretation and therefore includes uncertainties. Although the composition and configuration of engineered structures that can function as barriers in a geologic disposal facility can be defined with a degree of precision not possible for natural barriers, it is recognized that except for a few anthropogenic and natural analogues, there is a limited experience base for assessing the performance of complex, engineered structures over periods longer than a few hundred years, and hence, there are uncertainties in these projections. These uncertainties can be addressed in some measure by requiring the use of a multiple barrier approach; specifically, an engineered barrier system is required in addition to the natural barrier system provided by the geologic setting.

The safety case design strategy encompasses those activities focused on ensuring that the natural and engineered barriers work in concert, and that where possible, the engineered barriers work to assist the geologic setting in meeting the performance objectives for the period following permanent closure. Initially, emphasis is placed upon the ability to contain the wastes by waste packages; this is known as the containment period. A waste package is composed of the waste form and any containers, shielding, packing, and absorbent materials immediately surrounding an individual waste container. The waste package will be designed to complement the character of the host rock formation and thus maintain its integrity for long periods of time, and thereafter fail gradually, assuring the continued longer-term attenuation of radionuclide releases. Following the containment period emphasis is placed upon the ability to achieve isolation of the wastes by virtue of the characteristics of the geologic repository. Isolation means inhibiting the transport of radioactive material so that amounts and concentrations of the materials entering the accessible environment will be limited. Waste forms of used nuclear fuel (consisting of ceramic-like uranium oxide, specifically UO_2) and high-level radioactive waste (e.g., borosilicate glass) are relatively stable in the environments to be expected in the salt, clay and granite geologic media selected for consideration, which contributes to isolation. The engineered barrier system works to control the release of radioactive material to the geologic setting and the geologic setting works to control the release of radioactive material to the accessible environment.

The types of design features that can be considered in this context include, but are not limited to: design for reducing the potential for deleterious rock movement or fracturing of overlying or surrounding rock; using excavation methods that will limit the potential for creating preferential pathways for groundwater to contact the waste packages or radionuclide migration to the accessible environment; and taking into account the predicted thermal and thermo-mechanical response of the host rock, the surrounding strata, and the groundwater system.

The safety assessment evaluations of repository performance include consideration of uncertainty in the behavior of the repository system, and the results thus reflect the capability of each of the barriers to cope with a variety of challenges (e.g., combinations of parameters leading to less favorable performance for individual barriers and combinations of barriers).

2.2.3.1.1 Features of Generic Engineered Barrier Systems

Categorization of the physical features or components of the engineered barrier system is fairly consistent among the various national programs. These features and their design will differ somewhat for different disposal concepts (waste form, rock type, and concept of operations) depending on the needed waste isolation performance of the engineered barriers, the expected performance of the natural barrier system, and interactions between engineered and natural features. However, a broad set of common features or engineered barrier system components can be defined. In the Joint EC/NEA Engineered Barrier System Project report (Bennett 2010), four main engineered barrier system components are: waste form, container/overpack, buffer/backfill, and "others." Bennett et al. (2006) defined the main functions, independent of repository concept and/or disposal environment, of these four primary engineered components as:

- The waste form is designed to provide a stable matrix that is resistant to leaching and gives slow rates of radionuclide release for the long term
- The container/overpack is designed to facilitate waste handling, emplacement and retrievability, and to provide containment for up to 1,000 years or longer depending on the waste type and disposal concept
- The buffer/backfill is designed to stabilize the repository excavations and the thermal-hydrologic-chemical-mechanical conditions, and to provide low permeability and/or diffusivity, and/or long-term retardation
- The other engineered components (e.g., seals) are designed to prevent releases via tunnels and shafts and to prevent access to the repository

Figure 2-1 shows these components schematically, along with the primary components for the natural system.

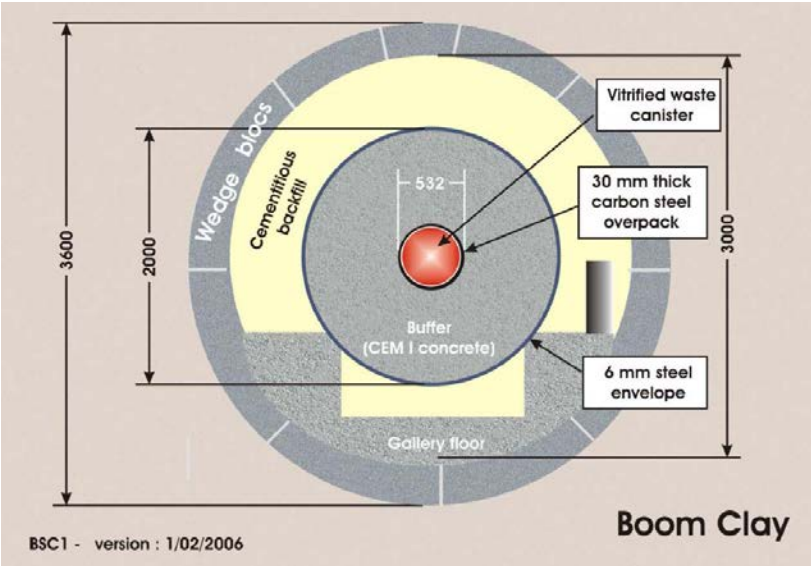
The composition of engineered barrier system components for various disposal concepts is presented in Table 2-3. The reference for a salt repository is the generic salt repository study (Carter et al. 2011a); the reference for a clay repository is the French concept (Andra 2005b); the reference for a granite repository is the Swedish KBS-3 concept (SKB 2006b; SKB 2011); and the reference for deep borehole disposal is the Sandia/MIT concept (Brady et al. 2009). It should be noted, however, that in parallel with the activities of the Joint EC/NEA Engineered Barrier System Project (Bennett 2010), other national programs have developed engineered barrier system designs in which the buffer/backfill "component" shown in Table 2-3 would be better represented as separate buffer and backfill components. For example, a more recent waste package concept is the "supercontainer" concept, selected in the Belgian program through a multi-attribute decision process (Bennett 2010). This waste package concept, shown in Figure 2-8, includes the buffer material within the "waste package" itself, where the "waste package" (called the "supercontainer") is comprised of a carbon steel "overpack", surrounded by a Portland cement concrete buffer, all contained in a stainless steel "envelope." The inner carbon steel overpack functions to prevent radionuclide releases during the initial thermal period; the concrete buffer provides a high pH environment to limit corrosion of the carbon steel overpack during this initial thermal phase; and the stainless steel envelope is for structural integrity and handling of the entire supercontainer. As shown in Figure 2-8, once emplaced, the entire supercontainer is surrounded by a cementitious backfill. As noted by Bennett (2010), the supercontainer idea is a recent example that emphasizes one of the key findings of the Joint EC/NEA Engineered Barrier System Project (Bennett 2010):

> The EBS is best regarded as a system of components *that functions in conjunction with the surrounding rock and thus provides acceptable levels of safety. The EBS should be tailored to the wastes that need to be disposed of, and to the host rock in which it is required to function. Each component of the EBS will have its own functions, but it is the functioning of the system as a whole that is most important. The importance of regarding the EBS as a* system *can be readily understood from examples in which the function of one EBS component is to protect a neighboring component.*

Table 2-3. Composition of Engineered Barrier Components for Various Disposal Concepts

Country/ Concept	Waste Form	Container/ Overpack	Buffer/Backfill	Others (seals, liner, etc.)
France; Clay[a]	Used UO_2 and mixed oxide (MOX) nuclear fuel	Stainless steel with metal insert	Bentonite buffer with metal disposal tube	Bentonite seals
	High-level radioactive waste borosilicate glass	Stainless steel container, steel overpack	Optional bentonite buffer	Bentonite seals
Sweden/KBS-3; Crystalline / Granite[a]	Used UO_2 nuclear fuel	Copper-iron	Bentonite	Tunnel backfill
U.S.; Generic Salt[b]	High-level radioactive waste glass	Stainless steel	Crushed/compacted salt	
U.S.; Deep Borehole[c]	Used UO_2 nuclear fuel and high-level radioactive waste glass	Carbon steel	Bentonite-water slurry around canister strings; compacted bentonite between canister strings; bentonite, cement, and crushed rock above disposal zone	Steel liner (casing)

NOTE: [a]Organisation for Economic Co-operation and Development 2003a.
[b]Carter et al. 2011a.
[c]Brady et al. 2009.



Source: Bennett 2010, Figure 3.5.

Figure 2-8. Belgian Supercontainer Engineered Barrier Concept

The following is a high-level description of the primary purpose(s) of each of the engineered barrier features or components shown in Table 2-3:

- **Waste Form—**The potential waste form matrix (e.g., ceramic, glass, glass-ceramic, metal) containing the radionuclides is a key component of the engineered barrier system. It must maintain stability and durability under long-term exposure to high radiation fields and corrosive environments, as well as provide dose mitigation associated with intrusion scenarios. The interaction of the waste form with the repository host rock, and with other engineered components, is an important issue for long-term disposal, and it may be appropriate to design some waste forms (e.g., from advanced fuel cycles) to accommodate specific repository environments. Waste forms such as high-level radioactive waste borosilicate glass have already been engineered for slow degradation rates and gradual release of radionuclides into the immediate environment. Used uranium oxide fuel has also been well-characterized, and degrades slowly in certain environments, such as under reducing conditions.
- **Cladding—**Cladding protects fuel from degradation in the reactor, and used fuel is likely to be received for disposal at the repository with cladding intact. Cladding can protect the used fuel from degradation in the repository also, especially in the event of a waste package breach during the period of elevated temperature, i.e., during the first few hundreds to thousands of years after emplacement. Cladding from commercial light-water reactors is generally made from Zircaloy (Type 1L is typical) a zirconium alloy that is chemically stable and resistant to corrosion. It can be damaged by internal pressurization of the fuel rods by fission-product gases; such damage can be controlled by limiting the fuel temperature.
- **Waste Containers, Packages, Overpacks, and Internal Features—**Corrosion resistant materials such as titanium, nickel-chromium alloys, etc. can provide very long containment lifetime for waste packages if oxidizing conditions are expected (containment is defined as no breach of any kind). Corrosion resistant materials generally are passive and subject to localized corrosion, which produces only small penetrations. By contrast, corrosion allowance materials such as copper and low-alloy steel are not subject to localized corrosion or similar degradation modes, but general corrosion is more rapid and breach could occur in 100,000 years or sooner in oxidizing conditions. However, in a reducing environment, corrosion allowance materials can last almost indefinitely. Modes of degradation for corrosion allowance materials are relatively few, and well understood. Corrosion allowance materials can be used to protect the package contents during the period of elevated temperature, gamma radiolysis, etc.

  The geometry of waste packages is typically cylindrical to facilitate handling, structural integrity, and economical fabrication and use of material (thin-walled canisters, and disposal overpacks as needed). Available fabrication and treatment methods include various welding techniques, friction welding, thermal or plasma spraying, annealing, burnishing, peening, etc. Different methods can be used within the same waste package assembly where materials and functions differ, for example, the inner canister may be closed differently than a disposal overpack. Other process steps may be taken during packaging (e.g., dewatering, charging with inert gas) to limit corrosion and promote heat transfer.

  Within the package, internal racks or inserts made of cast iron or other corroding materials, which initially have structural support functions, later can control the internal chemical environment around the package contents (i.e., used nuclear fuel or high-level radioactive waste). This can limit the rate of degradation of the waste form, and/or limit mobility of released radionuclides. In addition, “getter” materials may be added (no structural function) either inside or outside the waste package, to sequester released radionuclides directly or on contact with degradation products. In addition, used fuel containers contain internal features (flux traps, neutron absorbers) that prevent criticality during storage and transportation, and that will limit postclosure criticality also either in intact or degraded configurations.

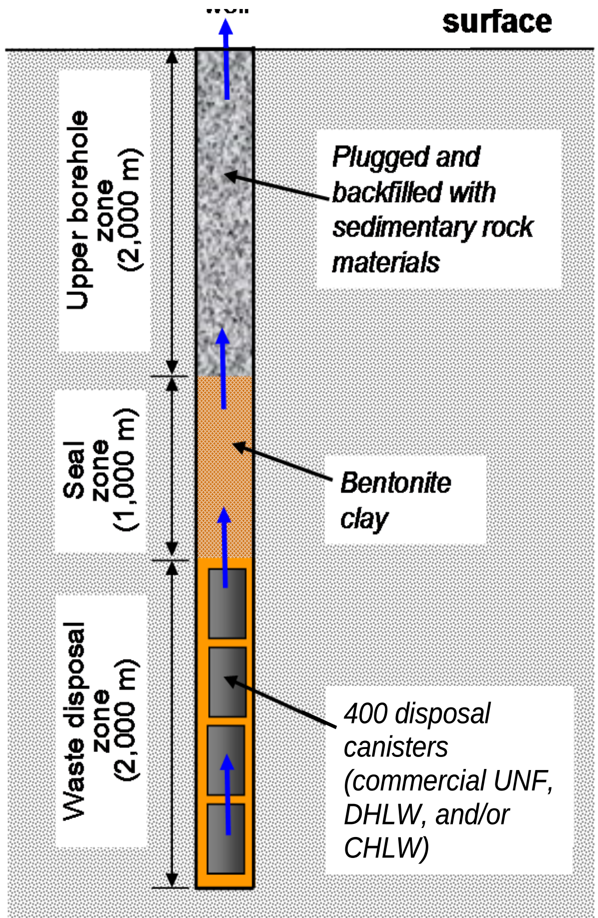
- **Buffer/Backfill—**Features that control the waste package corrosion environment include bentonite clay (e.g., the Swedish KBS-3 concept) or concrete (e.g., the Belgian supercontainer) buffers that are directly in contact with the waste package or other engineered components, and limit the availability of moisture, oxygen, or other potentially corrosive species. Similarly, crushed salt backfill within a salt repository isolates the waste packages and limits availability of corrosive species after creep reconsolidation transforms the backfill properties back to those of the native host rock.

  Backfilling of mined repository openings is used in saturated hydrogeologic settings to inhibit preferential groundwater flow into or out of the repository, and may also be used where no flow is expected. This contributes to the robustness aspect of multiple barriers, by serving to reduce uncertainty in predicting coupled hydro-mechanical responses (e.g., emplacement alcoves in the generic salt repository). In the disposal concepts developed in other countries for granite and clay media, low-permeability backfill is used to limit advective transport of water and radionuclides along access drifts proximal to emplaced wastes, and in smaller diameter boreholes. Backfill may also be used to provide swelling pressure to contain the swelling buffer in adjoining emplacement holes, and/or to exert pressure on the DRZ, limiting its permeability. Backfill materials such as those containing clay, may have the additional benefit in that they retard radionuclide transport.

- **Other Features—**The three primary capabilities of engineered barrier components that help enhance waste isolation performance are (U.S. Department of Energy 2008b): (1) limit water contacting waste, (2) limit the rate of radionuclide releases from the waste form, and (3) limit transport of radionuclides to the natural barrier system. In most repository concepts the waste package is the primary engineered feature that fulfills these capabilities, but other engineered barrier features/components may also be emplaced to provide defense-in-depth and to function in concert with the waste package. Some of these components include water diversion features, such as capillary barriers, which limit advective flow and transport. Tunnel liners can also serve in this role but usually are not as robust and long-lasting, and often only play a role in preclosure safety. Also, in open emplacement concepts an engineered pedestal may support each waste package. This also can serve to "isolate" the waste package from the host rock environment.

  A much more common engineered feature is some sort of plug or seal, which has a primary function of limiting access to the repository after closure. Plugging and sealing of access tunnels, shafts, and ramps also provides barriers to preferential liquid flow into or out of the repository. Thus the repository can occupy a low permeability interval within a geologic section that includes more permeable units above or below, while assuring isolation. A seal zone or sealing system is also important in the deep borehole concept, as shown in Figure 2-9. Its main function is to limit entry of water and migration of contaminants through the borehole after it is decommissioned, i.e., to isolate the emplaced wastes from the accessible environment. As indicated by Brady et al. (2009), the key characteristics of the seal system design are that it exhibits excellent durability and performance and is constructed with multiple, low-permeability materials to reduce uncertainty in performance. Bentonite is envisioned as a buffer/seal material because of its low permeability, high sorption capacity, self-sealing characteristics, and durability. Cement borehole plugs form an additional component of the deep borehole disposal seal system.

These features, and the processes that act upon them, affect the behavior of the engineered barrier system. Relevant processes in the engineered barriers are discussed further in Section 3.4.2. Functions of the engineered barrier features/components are discussed in Section 2.2.3.1.1.1.



NOTE: CHLW = commercial high-level radioactive waste, DHLW = U.S. Department of Energy high-level radioactive waste, UNF = used nuclear fuel

Source: Clayton et al. 2011, Figure 3.4-1.

Figure 2-9. Schematic Illustration of a Deep Borehole Disposal System

2.2.3.1.1.1 Functions of Engineered Barrier System Components for Each Disposal Environment

To achieve the intended level of performance and fulfill safety functions, the features and components of the engineered barrier system must be designed to work in complementary and synergistic fashion to the natural environment in which they are emplaced. As well, they must be designed to work together as a system, wherein each engineered component promotes the functioning of other components. An example of these synergistic functions was given above in regard to the concrete buffer and steel overpack in the Belgian supercontainer concept. The primary safety functions of engineered barrier system features and components have already been discussed above: isolation, containment, and limited and/or delayed releases. These primary functions are broken down into several “subfunctions” by some repository programs (Bailey et al. 2011). For example, the “limited and/or delayed releases” primary function is subdivided in the Belgian program as: limitation of releases, limitation of water flow, and retardation.

Table 2-4 provides a description of the high-level role or specific safety function(s) of each of the primary engineered barrier system components, for each of the four generic concepts: salt, clay, granite, and deep borehole. The specific functions listed are all part of the three primary functions of isolation, containment, and limited and/or delayed releases.

Table 2-4. Primary Safety Functions of Engineered Barrier Components for Various Disposal Concepts

Country / Concept	Waste Form	Container / Overpack	Buffer / Backfill	Others (seals, liner, etc.)
France; Clay[a]	Expected to provide resistance to leaching for 100,000 years in all scenarios	Expected to facilitate waste emplacement and retrieval, protect the waste from water and limit releases over the long term in all scenarios	Expected to control the thermal-hydrologic-mechanical environment and limit releases	The tunnel lining supports the host rock during the operational phase. The tunnel backfill and seals prevent access to the repository, prevent radionuclide transport along tunnels and hold the buffer in place
Sweden/KBS-3; Crystalline / Granite[a]	Expected to slow the rate of radionuclide release	Expected to provide radionuclide isolation for ~1,000,000 years in all scenarios	Expected to provide mechanical protection and act as a diffusional barrier for ~1,000,000 years in all scenarios	The tunnel backfill is expected to provide mechanical support for the buffer and the host rock for ~1,000,000 years in all scenarios
U.S.; Generic Salt[b]	Expected to be stable in a salt environment and provide containment/ slow release of radionuclides for tens of thousands of years	Primarily a preclosure safety function for handling and shielding (but a temperature limit is imposed to ensure the containers remain intact until after closure	Expected to prevent water contact with the waste (and therefore limit releases) after crushed salt reconsolidation; also for preclosure radiation shielding;	Shaft and drift seals prevent intrusion of brine from overlying, water-bearing strata
U.S.; Deep Borehole[c]	No credit taken as a flow or transport barrier	No credit taken as a flow or transport barrier after closure; serves only for handling and emplacement	Expected to limit migration of contaminants through the borehole via low permeability and high sorption capacity of bentonite	Seal system expected to limit entry of water and migration of contaminants through the borehole via low permeability and high sorption capacity of bentonite

NOTE: [a]Organisation for Economic Co-operation and Development 2003a.
[b]Carter et al. 2011a.
[c]Brady et al. 2009.

A final point is that even partial performance of degraded engineered barrier components can play an important role or safety function. Specifically, engineered barrier components are carefully designed and tested to serve specific functions that are measured in terms of time to failure. However, even after partial failures such as small localized corrosion penetration of a waste package, degradation of the waste form and release of radionuclides will occur very slowly. Hence the engineered barrier system continues to provide waste isolation performance even after initial failures occur, and this partial function should be represented in safety assessments using mechanistic models with an appropriate degree of process fidelity and spatial resolution.

2.2.3.2 Disposal Concepts

Disposal concepts for used nuclear fuel and high-level radioactive waste have been developed for various geologic media, building on the particular natural, waste isolation characteristics of those media, and identifying engineered barrier features that would be used. A disposal concept generally consists of a waste inventory for disposal, a rock type to be excavated, and an engineering concept of operations. The concept of operations recognizes specific contributions from natural and engineered barriers, and includes a description of the construction and operations needed for implementation. Disposal concepts have been developed iteratively, with many improvements, since geologic disposal for high-activity waste was first proposed (National Academy of Sciences 1957). In the U.S., concepts were developed for the Salt Repository Project (U.S. Department of Energy 1987), and tuff (U.S. Department of Energy 2008b), among others. Programs in other countries, such as those in Sweden, Finland, and France, have developed disposal concepts to a degree of maturity ready for implementation.

Mined geologic disposal concepts involve facilities with ramp or shaft access to the waste emplacement area; these facilities are typically planned for depths of several hundreds of meters. Ramps and shafts provide access for handling equipment and construction and operational personnel. The following emplacement modes for used nuclear fuel and solid, high-level radioactive waste in mined repositories have been identified by previous and ongoing geologic disposal investigations internationally and in the U.S.: in-drift emplacement; emplacement in vertical or horizontal boreholes drilled from mined openings; and backfilled alcove emplacement.

The plans for drifts, alcoves, and/or boreholes for waste emplacement, other tunnels for access and ventilation, and the shafts and ramps that provide access from the surface, are collectively referred to as the repository layout. The layout must accommodate the geologic structure, which is simple for extensive horizontal strata (e.g., bedded clay/shale or salt) and more intricate for geologic structures. Headings for all openings in the repository are considered with respect to trends in frequency and orientation of pre-existing fractures, which influence the engineering measures used to obtain the needed degree of long-term stability. By contrast, layout orientation in bedded salt could be selected for convenience, with no significant fracturing in the host rock. Ground support measures for intersections between openings are also considered in selecting layout orientation.

Layouts are designed for ease of access, construction, ventilation, operations, and closure. Ventilation systems are engineered to ensure separation of active work areas from radiological controlled areas, for example, using negative ventilation pressure in emplacement drifts. Repositories like any underground construction will have a drain plan whereby any groundwater inflow can be collected for removal from the facility. Repository layouts should be designed for pilot-scale development, so part of the repository is fully developed and waste emplaced, for evaluation of the disposal concept prior to further development. Layouts are also designed for eventual plugging and sealing at repository closure, for example, the separated disposal “panels” in the French concept can be sealed off independently in the future as phases of disposal are completed (Andra 2005b). This flexibility could facilitate reversibility or modification of the disposal concept, should nuclear waste management policy change in the future.

Spacings between waste packages or between emplacement boreholes, alcoves, or drifts can be adjusted to optimize thermal response. Different types of waste can be emplaced in separate panels, or parts of the repository, to optimize the layout along with related measures such as ventilation to remove heat, for thermal management. Panels can also be used for separate disposal of different waste types, and to facilitate waste retrieval, if necessary, or repository closure in the future. Access from the surface can be provided by shafts, ramps, or boreholes or combinations selected for optimal movement of ventilation air, workers and materials, waste rock, and waste packages. Vertical layouts (e.g., deep boreholes) have been proposed, as have multi-level layouts (e.g., to follow a salt diapir). An array of vertical boreholes is the simplest design for deep borehole disposal, although branching, inclined boreholes have also been suggested. Excavation and construction methods needed for repository development are well within the state-of-the-art in underground construction.

Nuclear waste disposal in very deep boreholes drilled from the ground surface also has been investigated in the U.S. and internationally for many years (Section 1.4). Deep disposal boreholes would be drilled to approximately 5-km depth (Figure 2-9) and spaced approximately 200 m apart to limit thermal interactions and to allow for some borehole deviation. Deep borehole disposal could have an advantage in thermal management because waste canisters would be small and contain relatively small amounts of heat-generating waste, and emplacement boreholes would be widely spaced. Isolation performance would be provided mainly by the far-field host medium (crystalline basement rock) and the long sealed interval of the borehole above the waste. Suitability of the host medium can be determined using established methods for geophysical, geochemical, and hydrologic measurements in wells. Borehole disposal is well suited for low-volume, solid waste forms containing long-lived radionuclides, where additional assurance of waste isolation is desired, or to limit the scope of a companion mined repository for other waste types. Retrieval of emplaced wastes from a deep borehole would be expected to be much more difficult than from mined geologic disposal.

2.2.3.2.1 Open vs. Enclosed Emplacement Modes

Mined disposal concepts, in which subsurface access is provided by a shaft or ramp, can be identified using combinations of four key attributes: competent/plastic host rock, high/low permeability host rock, saturated/unsaturated hydrogeologic setting, and whether a low-permeability backfill or buffer is installed around waste packages prior to closure. If waste packages are emplaced directly against the host rock, or a borehole or drift liner, or against an engineered layer such as a clay buffer, then the concept is an enclosed emplacement mode. If waste packages are emplaced in larger air spaces and can be ventilated, for example to remove heat or moisture, then the concept is an open emplacement mode. Enclosed emplacement modes typically involve temperature limits on the contacting material, whether it is clay buffer material, natural clay-based rock, rock salt, etc. Therefore, enclosed mode designs typically involve smaller capacity waste packages (e.g., 4 pressurized water assemblies per package) (Hardin et al. 2012). Open emplacement modes can be used to emplace hotter waste and/or larger capacity waste packages (e.g., 21 pressurized water assemblies per package (Hardin et al. 2012)), while still meeting temperature limits, by using ventilation for heat removal initially, when the waste heat generation rate is greatest. Note that low permeability backfill (which could include a buffer around waste packages) can be installed at the time of waste emplacement (an enclosed mode) or the repository can be ventilated for a time and the backfill installed later (an open mode).

2.2.3.2.2 Key Attributes for Mined Disposal Concepts

A basic distinction is whether the host medium is “plastic” (self-sealing and openings are prone to collapse) or “competent” (fracture permeability persists and openings are unlikely to collapse) on the timescale of a few tens of years. Open modes require that ventilated emplacement openings remain open and stable during repository operations, at least until thermal limits can be met if the openings collapse. Both open- and enclosed-mode disposal concepts in low-permeability plastic host media rely on deformation to seal fractures. These are working definitions that may also include rock characteristics such as the type of in-situ stress condition and its magnitude, and the extent of excavation damage.

Another key attribute is the permeability of the host geologic formation. Low-permeability host media (less than 10^{-16} m^2) generally act as confining units in hydrogeologic systems, so that natural or induced groundwater flow rates within these units are small. One consequence is that low-permeability formations are often reducing if chemical oxygen demand (i.e., minerals, organic matter) is present.

Water saturation of the host rock is another key attribute, but is only important for higher permeability media. In low permeability media (less than 10^{-16} m^2) water may be present but does not flow at rates significant to waste isolation, whether the formation is nominally saturated or unsaturated. Whereas disposal concepts for used nuclear fuel and high-level radioactive waste being investigated internationally are intended for saturated settings, disposal of hazardous waste, low-level waste, and even municipal refuse is typically done in unsaturated settings, using barriers such as caps and liners that are designed to maintain unsaturated conditions around the waste.

A final key attribute is whether engineered, low-permeability buffer and/or backfill material is installed around the waste packages prior to closure. Disposal in low-permeability media such as clay or salt, could allow simpler waste packaging, without additional backfilling of the emplacement openings at closure, with excellent isolation performance. These concepts rely mainly on natural barriers for waste isolation, whereas others (e.g., KBS-3 concept; SKBF/KBS 1983) rely more on engineered barriers to achieve required performance.

Based on this discussion, key attributes for describing disposal concepts may be summarized:

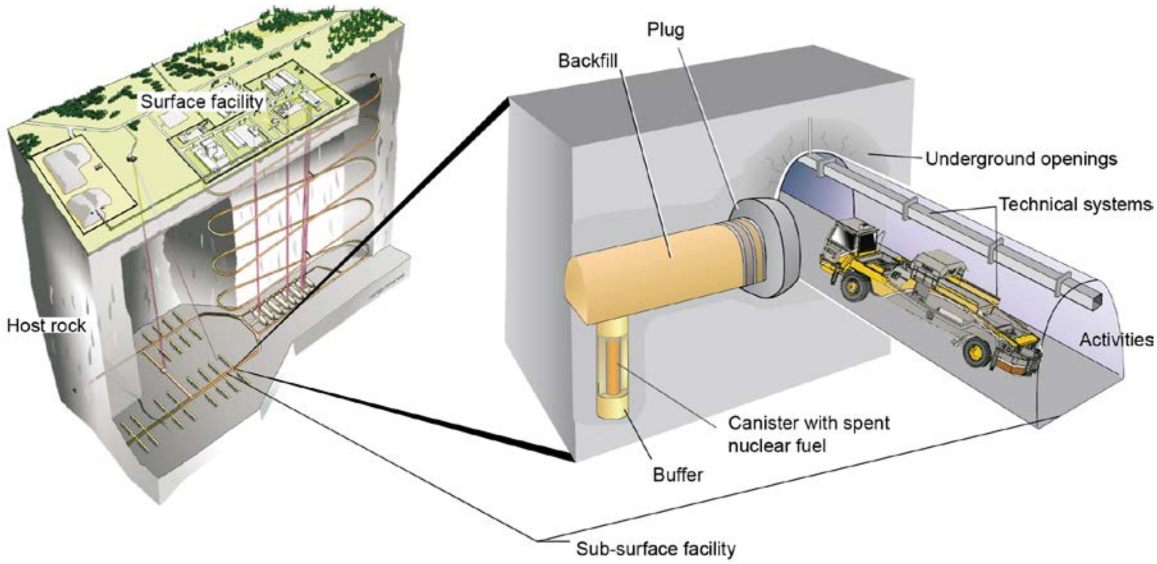
- Plastic vs. competent host rock (persistence of fracture permeability, and stability of underground openings in heated rock on timescales of years to tens of years)
- Low-permeability vs. higher permeability host rock (importance of host rock performance vs. engineered material performance to the isolation performance of the disposal system)
- Saturated vs. unsaturated (types of barriers needed to isolate waste from groundwater, potential for saturated flow paths within the repository particularly after permanent closure, likelihood of reducing vs. oxidizing chemical conditions)
- Open after closure vs. installation of low-permeability backfill at closure (whether low-permeability backfill and plugs are needed to control groundwater flow in the repository after permanent closure)

These attributes were used in selecting a simple set of reference disposal concepts that is well suited to guiding generic research and development activities, and enabling future disposal site evaluations and selection for the broad range of geologic settings available in the U.S.

A number of reference disposal concepts are described in the following subsections. These disposal concepts emphasize the disposal of waste packages containing used nuclear fuel. While these disposal concepts have been evaluated for disposal of high-level radioactive waste (Hardin et al. 2012), cost and thermal considerations suggest that the design of the disposal concepts is likely to be more sensitive to how waste packages containing used nuclear fuel are arrayed. Accordingly, the disposal of waste packages containing used nuclear fuel is emphasized.

2.2.3.2.3 Crystalline Rock Enclosed Mode Concept

The crystalline rock disposal concept has been extensively investigated in Sweden and Finland, where repository development is planned for the next decade. A conceptual illustration of the KBS-3 repository design concept is shown in Figure 2-10. For consistency with those concepts the reference crystalline rock repository depth is assumed to be 500 m below the surface, in hydrologically saturated granitic host rock with limited permeability, in which hydraulic flow potential gradients are very small. These conditions are expected to result in very slow groundwater flow typical of the Canadian and Baltic Shields, and the host rock chemical environment is expected to be reducing. The subsurface layout consists of parallel emplacement drifts spaced approximately 20 m apart, with waste packages emplaced in vertical boreholes drilled into the floor from these drifts. The parallel access drifts have sufficient diameter to provide clearance for drilling equipment and waste package handling. A single package will be emplaced in each borehole, with boreholes spaced 6 to 10 m apart. Waste packages are thick-walled, made from copper or carbon steel (a choice to be made based on economics and performance) with welded closures. The space between the package and the emplacement borehole wall (typically 35 cm) is filled with a low-permeability buffer material consisting of swelling clay (e.g., sodium bentonite) emplaced initially in dry, compacted form. The buffer swells on contact with groundwater, effectively sealing the waste package from direct contact with groundwater. Construction may be expedited by use of prefabricated assemblies consisting of a single waste package and the surrounding clay buffer in compacted dry form, held together by a steel envelope (McKinley et al. 2006).



Source: SKB 2006b.

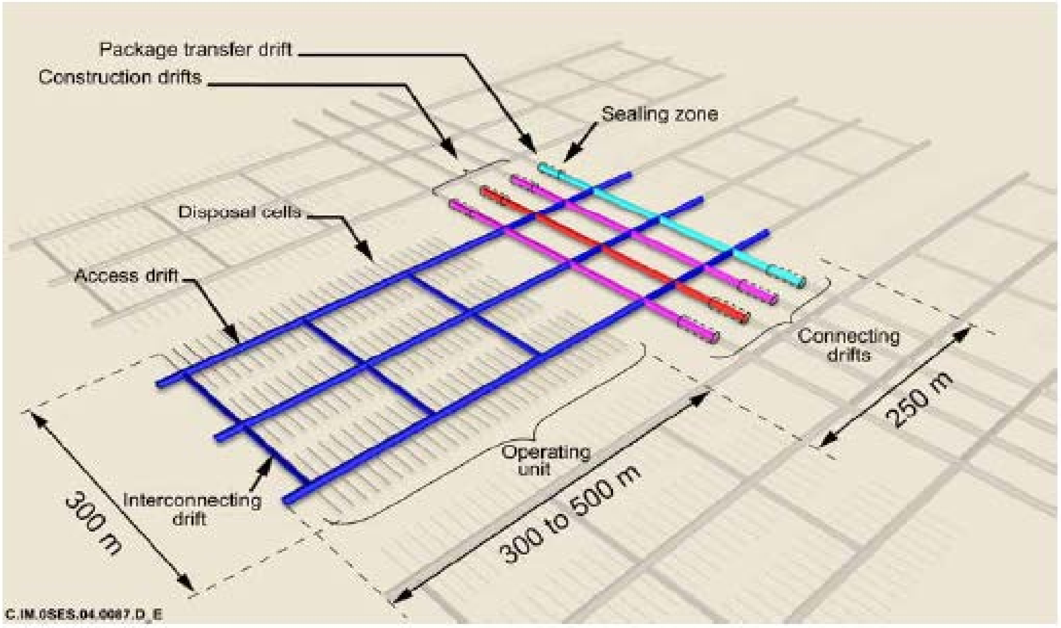
Figure 2-10. Conceptual Illustration of the KBS-3 Repository Design Concept for Crystalline Rock

2.2.3.2.4 Clay Enclosed Mode Concept

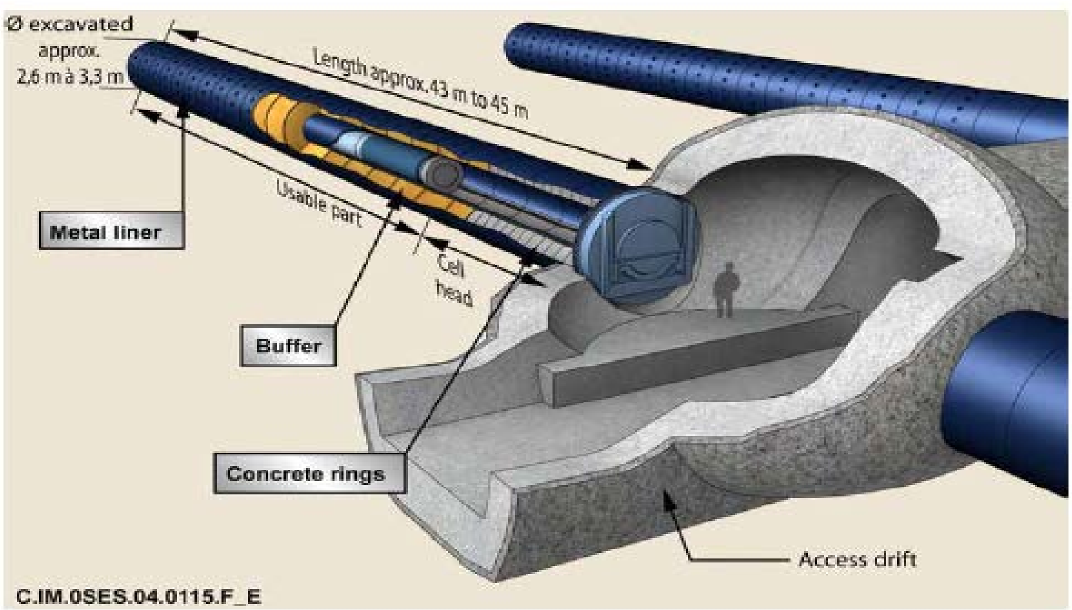
A conceptual illustration of a repository design concept for clay is shown in Figure 2-11(a), and details of the emplacement borehole are shown in Figure 2-11(b).

For the reference clay disposal concept, low-permeability clay sediments are assumed to be 150-m thick, situated so that a suitable interval for repository development exists at a depth of 500 m (similar to the stratigraphy at the Bure location in France). Tunnels and drifts will be excavated using mechanized mining equipment. Horizontal emplacement is preferred over vertical, to accommodate limited thickness of preferred layers for emplacement within the clay sequence. Accordingly, the reference concept for used nuclear fuel disposal makes use of in-drift emplacement directly in horizontal drifts. Stainless steel canisters containing used nuclear fuel will be inserted into carbon steel overpacks, and placed horizontally in steel-lined, small diameter emplacement drifts, surrounded by bentonite buffer material. The reference waste package spacing is 10 m for in-drift emplacement of used nuclear fuel (packages are nominally 5-m long), and drifts are spaced 30 m apart. These dimensions are comparable to those proposed for the French repository (Andra 2005b) but with larger inter-package spacings to allow for hotter used nuclear fuel. Access drifts have sufficient diameter to provide clearance for drilling equipment and waste package handling. Plugs and seals at the collar of each emplacement drift will limit dessication during repository operations, provide radiation shielding, and inhibit movement of radionuclides into the access drift openings after repository closure. Access drift openings will be backfilled at closure using mined clay material processed to enhance low permeability and swelling potential.

(a)



(b)

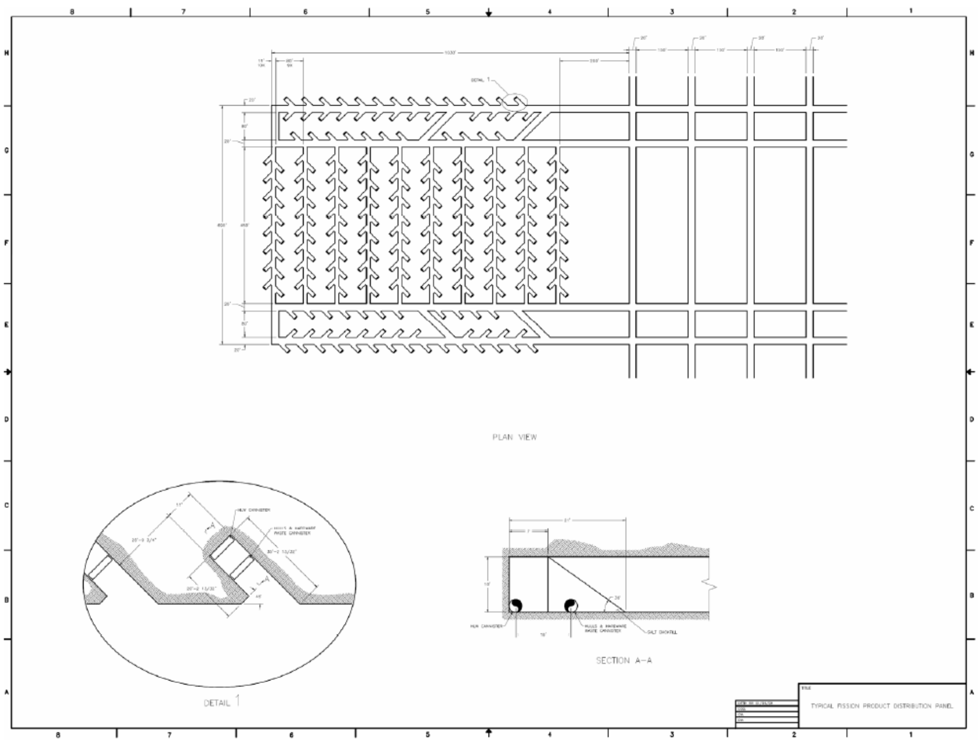


Source: Andra 2005c.

Figure 2-11. Conceptual Illustration of (a) a Repository Design Concept for Clay, and (b) Details of the Emplacement Borehole

2.2.3.2.5 Generic Salt Repository Concept

The salt repository concept is a geometrically simple disposal scheme whereby each used nuclear fuel waste package is placed on the floor, at the back of a mined alcove. Canisters used in storage and transport are inserted in steel overpacks with welded closures. Waste packages are immediately covered with crushed salt from repository excavation, to provide radiation shielding and to facilitate eventual creep closure and resealing of repository openings. A conceptual layout for a generic salt repository is shown in Figure 2-12.



Source: Carter et al. 2011a.

Figure 2-12. Conceptual Layout for a Generic Salt Repository

The original generic salt repository study (Carter et al. 2011a) generated some implementing principles including: (1) bedded salt is preferred over domal salt because it typically has much greater lateral extent; (2) use ground support and other repository features similar to WIPP; (3) use rubber-tire vehicles for construction and disposal operations to limit construction costs and increase operational flexibility; (4) avoid use of large diameter emplacement holes which are relatively expensive and tend to complicate waste emplacement; (5) use unshielded waste packages to reduce size, weight and cost; and (6) use narrow room widths to improve mining efficiency and structural stability. Although previous conceptual designs for repositories in salt called for emplacing waste packages in large diameter, vertical or horizontal boreholes (U.S. Department of Energy 1987), experience at WIPP has shown that construction of such boreholes presents challenges relative to in-drift emplacement (Pecos Management Services 2010). Pre-drilled horizontal disposal boreholes require wide room spans, and vertical holes in the floor require high overhead clearance, to accommodate drilling and emplacement equipment. Note that borehole emplacement as proposed in the Deaf Smith repository concept (U.S. Department of Energy 1987) could be adopted if needed, for example, to promote heat transfer with the intact salt.

2.2.3.2.6 Shale Open Emplacement Mode Concept

The shale open emplacement mode concept for used nuclear fuel disposal is similar to the clay enclosed mode described above, but without clay-based engineered buffer material around the waste packages. Emplacement drifts or segments of drifts will be plugged and sealed after the ventilation period and prior to repository closure, isolating these segments and preventing the potential for advective water flow throughout the repository. Ventilation will be used for as long as needed, e.g., for heat removal. Opening stability and protection of the shale medium from desiccation during operations will be controlled using a liner such as a thin-walled steel tube, or shotcrete. A conceptual illustration of an open mode geologic disposal facility concept for clay is shown in Figure 2-13.

After the cessation of ventilation or after repository closure, the drift segments will gradually fill with rubble while isolated from the rest of the repository by plugs and seals. Use of plugs or seals at the ends of each emplacement drift, without complete backfilling at closure, is suggested by the repository compartmentalization proposed by the French program (Andra 2005b, Section 2.3.1), and by the waste isolation strategy represented in safety assessment models (Clayton et al. 2011). Radionuclide mass flux along each transport pathway depends initially on the source concentration and not the mass of radionuclides available at the source (some radionuclides can eventually be depleted at the source by transport). For diffusion-dominated transport, waste isolation performance can be assured without the additional expense, operational risk, and worker dose associated with backfilling emplacement drifts at closure.

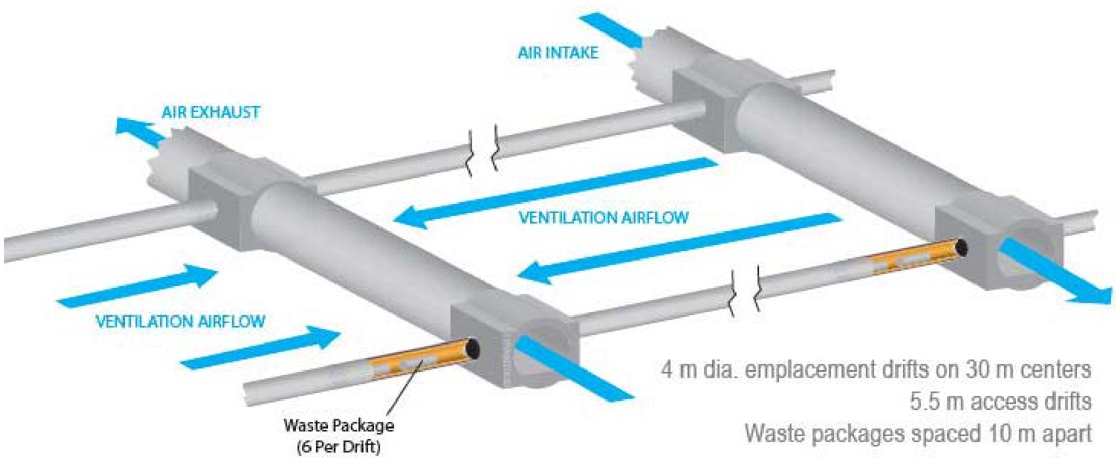


Figure 2-13 Conceptual Illustration of an Open-Mode Geologic Disposal Facility Concept for Clay

2.2.3.2.7 Backfilled Open Emplacement Mode Concept

The concept of operations is similar to the shale open-mode concept, which uses in-drift emplacement. Ground support consists of an initial lining of reinforced shotcrete, supplemented as necessary with steel ribs or girders and additional shotcrete, depending on rock characteristics and construction experience (U.S. Department of Transportation 2009). The underground repository would be accessed with vertical shafts and an inclined ramp for waste handling. Canisters containing used nuclear fuel will be inserted into disposal overpacks, and emplaced directly in dedicated drifts. The overpack will provide handling and structural support, and containment for some limited duration to support waste isolation performance objectives. Repository openings are backfilled before closure with low permeability, granular material engineered to impose a diffusion dominated, sorptive barrier to radionuclide release. Emplacement drifts or drift segments will be relatively short, accommodating only a few waste packages, to facilitate access for backfilling at closure (e.g., by remote auger delivery of granular backfill material). A conceptual illustration of the backfilled open mode geologic disposal facility concept is shown in Figure 2-14.

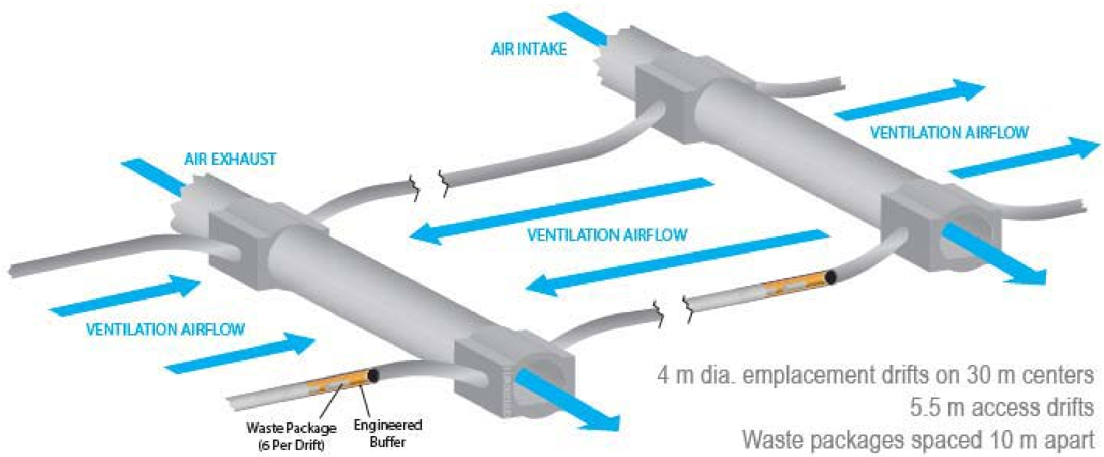


Figure 2-14. Conceptual Illustration of a Backfilled Open-Mode Geologic Disposal Facility Concept

2.2.3.2.8 "Hybrid" Salt Repository Concept

This is an extension of the generic salt repository described above, with the addition of dedicated drifts to remove heat by forced ventilation. Salt has unique properties for thermal management because of its high thermal diffusivity and resistance to degradation from temperatures on the order of 200°C or higher. Salt is not suited to open emplacement modes with in-drift emplacement as discussed above, because the closure of underground openings accelerates at elevated temperature. Accordingly, to stay open for heat removal the ventilated openings must be maintained and set apart from the emplacement openings. The "hybrid" reference mode does this with a simple change to the generic salt repository concept by adding parallel ventilation drifts. The cross-section of the ventilation drifts is circular to enhance long-term stability. The duration of ventilation could be up to several decades, depending on cooling needs for the types of waste emplaced in the adjacent emplacement alcoves. Calculations have shown that this mode of heat removal can decrease peak salt temperature (contacting the waste packages) by approximately 50 C°. This result could also be achieved with longer decay storage of the waste before emplacement in the repository, but the "hybrid" mode allows emplacement decades sooner.

A conceptual illustration of a hybrid salt geologic disposal facility concept for hotter waste is shown in Figure 2-15.

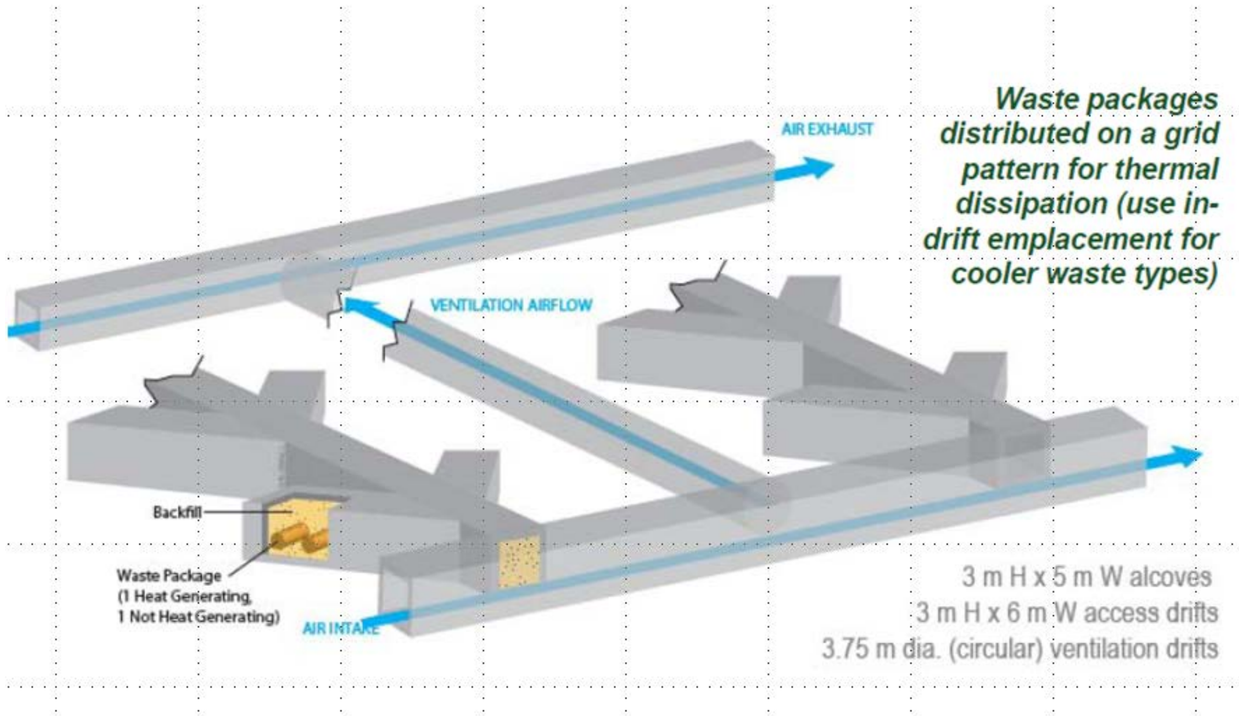


Figure 2-15. Conceptual Illustration of a Hybrid Salt Geologic Disposal Facility Concept for Hotter Waste

2.2.3.3 Thermal Constraints

Thermal management can be an important part of the strategy for geologic disposal of heat-generating waste. Disposal systems have the capability to dissipate thermal power up to limits determined by characteristics of the engineered barriers and natural materials in the repository near field. These limits may be reached depending on the physical form of the waste, and physical aspects of the disposal concept, in addition to the waste heat output. Thermal limits need not be controlling, particularly if the waste heat is limited and/or other aspects of the system control heat dissipation (e.g., host rock geology or repository operations). Waste heat output can always be reduced by long-term decay storage prior to disposal, with the additional cost of storage operations. The additional time and expense of surface decay storage prior to disposal makes thermal management a popular topic for discourse on geologic disposal. The following paragraphs summarize the types of thermal constraints and temperature limits, and the measures available for control.

2.2.3.3.1 Types of Thermal Constraints

Thermal constrains on repository configuration and waste emplacement may be imposed for various reasons: to limit boiling or thermally driven coupled thermal-hydrologic-chemical-mechanical processes, to limit peak temperatures for sensitive engineered or natural materials, and to limit time-temperature exposure and thereby slow rate-limited processes such as creep or de-alloying. Which materials and components are protected depends on the disposal concept and how waste isolation performance is allocated to the components.

Two types of temperature limits are generally applied in repository analyses: peak temperature and temperature-time exposure. Peak temperature limits are appropriate to prevent relatively rapid processes that exhibit temperature threshold-like behavior (e.g., cementation in clay buffers). Engineered materials may exhibit temperature-time dependent degradation, whereby a function of both time and temperature is most important (e.g., metal de-alloying). Other degradation processes may have rates that are functions of temperature and mechanical load (e.g. high temperature creep).

Thermal constraints may be imposed as limiting temperatures, or limiting results (minimal or maximal) for thermally driven processes. Such constraints may be imposed in the near field or far field, for engineered features or for the host rock (Hardin et al. 2011a).

Far-Field Thermal Constraints—The disposal concept, including the heat output of waste and how densely it is emplaced, can be adjusted to control the following:

- **Rock Stress and Displacement**—Limit thermally induced stresses or displacements in the host rock or adjacent units close to the repository, to limit formation of new flow paths, or to limit degradation of boreholes and mined openings. For some disposal concepts such as mined disposal in salt or clay, large rock deformations are desirable because they close openings and seal the repository to fluid movement. Thermal constraints may also be used to limit large-scale thermal expansion to control induced fracturing or displacement along existing faults or fractures.
- **Thermally Driven Coupled Processes**— Limit thermally driven coupled processes, for example, large-scale thermal-hydrologic-mechanical processes in clay, which are associated with thermal expansion of pore fluid. Such processes may be sensitive to threshold temperature effects, such as the boiling point of water, or they may be sensitive to thermal gradients. Another example is a mid-pillar temperature limit, which may be used to help ensure that liquid water in the host rock could readily drain vertically through the repository horizon.

While these types of constraints pertain to the far field, they also pertain to the near field where temperatures are greater. Constraining the peak temperature and the temperature history in the near field can effectively control temperatures in the far field, for most disposal concepts. For most disposal concepts, near-field temperature limits have been found sufficient to limit far-field effects.

Near-Field/Engineered Barrier System Thermal Constraints—Thermal constraints for important disposal concepts included the following types:

- **Alteration of Engineered Materials**— Near-field temperature is limited to control alteration of engineered barrier materials, especially clay buffers. Swelling clay can degrade by cementation, which typically involves partial dissolution of the clay, aqueous transport, and precipitation of silica or other secondary phases. Clay alteration generally degrades swelling pressure, increases rigidity promoting fracture, and potentially decreases sorption. Cementation processes are accelerated when water evaporates or boils, so temperature limits are used to limit clay degradation in the presence of moisture.
- **Host Rock Chemical Alteration**— Limit mineralogical changes in clay host media. Natural clay or shale formations typically contain more impurities, such as potassium, that can react with clay minerals, so temperature limits may be lower than for clay buffers.
- **Host Rock Mechanical Degradation**— Limit thermally induced micro-cracking in the less ductile crystalline rock types (e.g., igneous or metamorphic) to avoid degradation of in-situ mechanical properties or creation of flow paths.

Limiting the waste package surface temperature can be used to limit the peak temperature anywhere in the engineered barrier system or near field outside the waste package (Hardin et al. 2011a). This is often appropriate because the waste package and its contents can typically withstand higher temperatures than the surrounding engineered or natural materials. Temperature effects on the surface environment could also be important; they would be addressed in an environmental impact statement.

The foregoing list is representative but not a complete treatment of thermal effects on features and processes that control repository performance. The systematic framework for safety assessment, developed from years of experience (Bonano et al. 2010) is based on a complete evaluation of all features, events and processes that could affect performance, including thermal effects.

2.2.3.3.2 Temperature Limits for Disposal Concepts

Specific temperature limits and other thermal constraints identified by Hardin et al. (2011a) for the four reference disposal concepts—salt, clay, granite, and deep borehole—are summarized here. Higher temperature limits are currently under investigation, with the goal to accommodate greater thermal loads while maintaining safety.

Temperature Limits for the Salt Repository Concept—For salt, a more ductile material, a target value of 200°C has been proposed for the maximum temperature to limit uncertainty in performance models, although higher peak temperatures may be possible if supported by test data (BMWi 2008). A prior investigation for disposal of high-activity waste in a salt repository indicated a maximum allowable repository temperature of 250°C (U.S. Department of Energy 1987). Higher peak temperatures may be possible since rock salt is generally stable to mineralogical changes and contains little moisture.

Temperature Limits for the Clay Concept—The basis for temperature limits for natural, clay-rich host rock is similar to that for clay buffers. For example, such a limit was adopted by the French program for near-field clay-rich host rock (80°C; Andra 2005b). Elevated temperature causes thermal expansion of pore fluids, greater mineral solubilities, and changes in flow and transport properties.

Note that in saturated settings where the repository is situated hundreds of meters below the nominal water table, the local boiling temperature for water may be well over 200°C. The engineered barrier system is unsaturated when peak temperatures occur soon after emplacement, especially in low-permeability host media and in repository openings that are initially unsaturated or dehydrated, and have been backfilled, plugged, and/or sealed to inhibit water ingress. Thus, boiling could occur locally within the engineered barrier system with temperatures near 100°C, and the European programs have adopted corresponding temperature limits.

Temperature Limits for the Granite Rock Concept—Limiting the peak temperature of clay-based buffer materials to 100°C or cooler can prevent degradation from chemical processes such as cementation, enhanced by multiphase circulation and heat/mass transfer. For example, the French authority Andra proposed a 90°C limit for the hottest point in swelling clay buffers, while the Swedish program has adopted a peak temperature of 100°C. Variations on clay buffer limits have been proposed, for example, limiting an outer portion of the buffer cross-section to 125°C (Nagra 2002).

The near-field environment is initially unsaturated at repository closure, with much air present in the pore spaces of the clay buffer, backfill, and other engineered and natural materials. Although a repository may reside below the nominal water table, initially dehydrated buffer material may not hydrate until well into or after the thermal period. Other studies promote prefabricated, or "supercontainer" engineered barrier system construction in which the dehydrated clay buffer is enclosed in a steel envelope that provides protection from moisture during the thermal period, and the associated coupled processes (McKinley at al. 2006). The steel envelope would degrade more quickly than the waste package, so that buffer hydration can occur (at lower temperature) prior to waste package degradation. Crystalline rock can withstand temperatures of 200°C or greater by itself without significant degradation, but with the use of clay buffers the clay temperature limit is controlling.

Temperature Limits for the Deep Borehole Concept—For the deep borehole disposal concept no waste package or near-field temperature limits have been recognized because no performance credit is taken for the package or the near-field host rock (Brady et al. 2009). Also, for the deep borehole disposal concept peak temperatures are relatively low (Hardin et al. 2011a), because the packages are small. Whereas peak temperatures occur in the near field, the borehole seals and the far-field natural barriers provide sufficient waste isolation capability, so that near-field temperature limits are not needed.

Engineered Material Temperature Exposure Limits—Thermal constraints for waste forms could apply to any disposal concept for which waste form integrity or slow degradation is important. For example, fuel cladding (Zircaloy) temperature is limited to 400°C for normal conditions of storage and emplacement (Siegmann 2006) and thermal constraints also exist for repeated heating/cooling cycles. Cladding temperature has been limited to 350°C during permanent disposal (BSC 2005a). Use of multiple limits for short-term normal conditions, off-normal conditions, and long-term disposal reflects the time-temperature behavior of cladding creep, and the accumulation of creep damage.

Another example is maintaining the temperature of high-level radioactive waste glass. The peak centerline temperature of borosilicate glass waste forms must be kept below 500°C at all times to avoid devitrification (crystallization; Lutze 2006). Devitrification also exhibits time-temperature behavior, but at temperatures above 500°C so that a simple temperature limit can be used. Temperature limits for lanthanide glass, glass-bonded zeolite, and metal alloy wastes from electro-chemical processing have also been developed (Carter and Luptak 2010).

In summary, thermal constraints have been developed to protect features of the engineered and natural barriers, where the condition of those features is important for waste isolation performance. Repository temperature fields are relatively simple to estimate, especially for media in which heat transfer is dominated by thermal conduction. Avoidance of higher temperatures (e.g., above boiling) can significantly reduce uncertainty and complexity in predictive models of repository performance, by limiting phenomenology and the effort needed for characterization, modeling, and licensing.

2.3 Assessment Strategy

2.3.1 Overview

A safety assessment is undertaken to analyze the behavior of a geologic disposal system and its component subsystems. This type of safety analysis provides a quantitative estimation of the ability of specific subsystems, in isolation or in conjunction with other subsystems, to limit or delay the release of radioactive contaminants to the environment accessible to humans. The waste isolation capability of subsystems and of the total system is determined by estimating the numerical value of specific system and subsystem performance measures.

Because of the large temporal and spatial scales required to analyze radioactive waste disposal systems (i.e., tens of kilometers and thousands to hundreds of thousands of years), the effects of uncertainties must be quantified and propagated through a safety assessment. To a large extent, the credibility of the analysis and its results depend on the manner in which uncertainties are identified and quantified.

A quantitative safety assessment is performed as part of an overarching safety assessment methodology. The safety assessment methodology has been developed over a period of 40 years for probabilistic risk analysis of radioactive waste disposal methods, facilities, and systems and has been used to inform key decisions concerning radioactive waste management both in the U.S. and internationally (Meacham et al. 2011, Section 1.1).

In the U.S., the term performance assessment is synonymous with the term safety assessment; both terms are used in this report. The following excerpts summarize the safety assessment methodology (referred to in the excerpts as the PA, or performance assessment, methodology) (Meacham et al. 2011, Section 1.2.1):

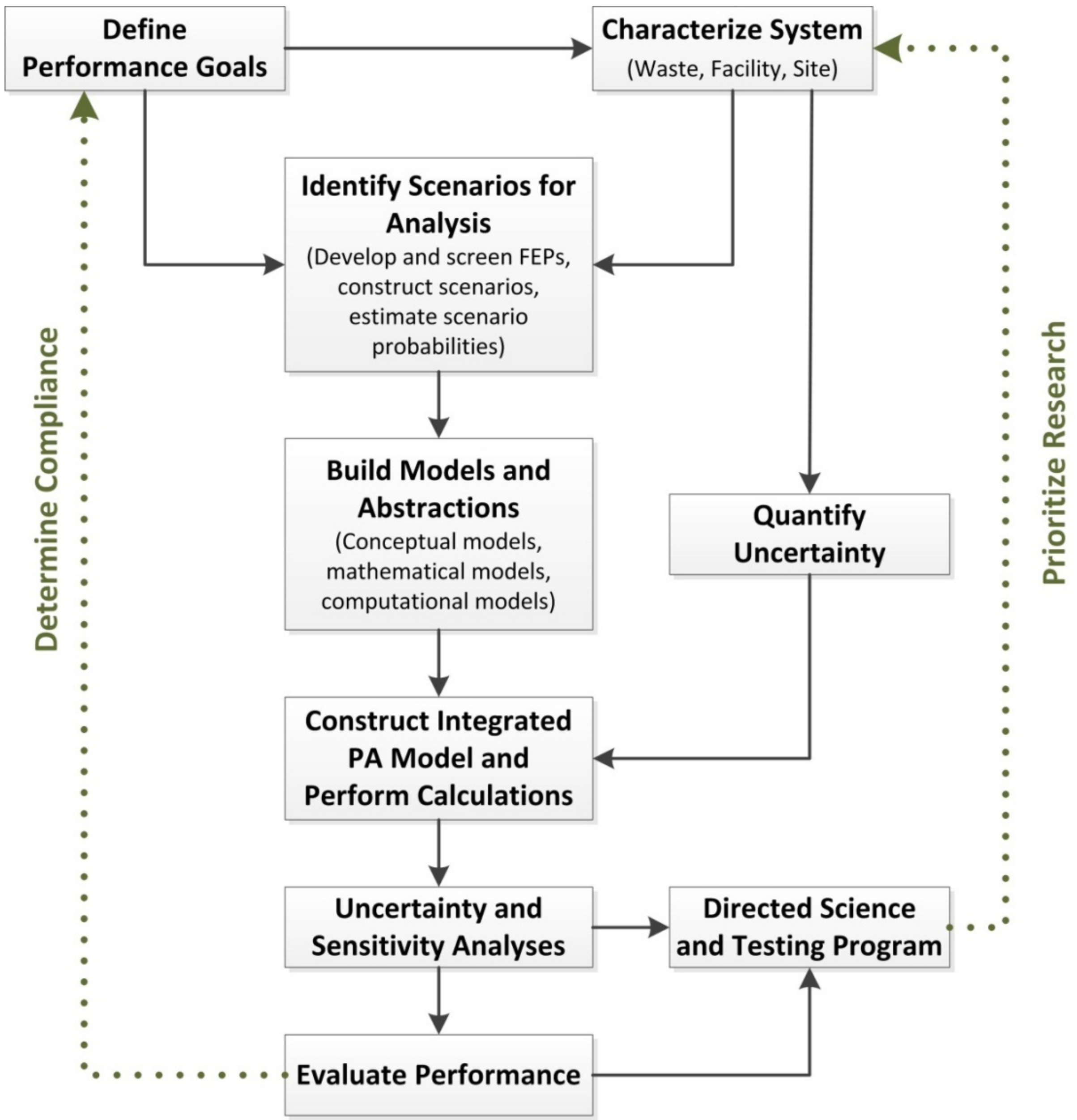
> *The PA methodology provides the framework for assembling, organizing, and assessing the large quantity of data and information needed to evaluate the performance of complex systems, such as radioactive waste disposal systems. The PA methodology incorporates data and information from multiple sources and organizes them in a logical manner to support decision making, explicitly taking into consideration the different sources of uncertainty that will influence the analysis. It also provides a framework that*

> *enhances the traceability, transparency, reproducibility, and retrievability of the technical work. Finally, it allows for the analysis of how the different components (i.e., subsystems) of the disposal system behave in isolation and in conjunction with each other.*
>
> *Radioactive waste disposal projects typically evolve over several decades, moving through several iterative phases during that timeframe, from early site selection, site characterization, preliminary analyses, and, eventually, the safety assessments employed to inform a final licensing decision for a disposal facility. The PA methodology has served as an effective management tool for such long term, evolutionary projects because the same methodology is used from initial research and development to the final safety assessment. The results of early applications of the methodology are used to systematically validate the system design, focus the associated research and testing program in the most important areas, identify opportunities to reduce costs, and ensure the design incorporates best practices.*
>
> *In the very early phase of a radioactive waste disposal project, applications of the PA methodology tend to be exploratory in nature and rely on relatively simple models focused on the identification of opportunities for improving understanding of the system under consideration. As the project evolves, more detailed models are incorporated into the methodology. In the intermediate phases of the project, applications of the methodology provide opportunities to review alternative models, conduct uncertainty and sensitivity analyses, identify shortcomings in the analysis or model implementation, and communicate with stakeholders. Eventually, once the understanding of the disposal system is sufficiently mature to proceed to the licensing phase, the application of the PA methodology provides the foundation for the safety analysis that informs the licensing decision.*

The safety assessment methodology is fundamentally the same for all project phases and generally comprises the following steps (Meacham et al. 2011, Section 1.2.2):

1. Define performance goals
2. Characterize system (waste, facility and site)
3. Identify scenarios for analysis
   a. Identify and screen potentially relevant features, events, and processes (FEPs)
   b. Construct and screen scenarios
   c. Estimate scenario probabilities
3.
4. Build models and abstractions
   a. Conceptual models
   b. Mathematical models
   c. Computational models
5. Quantify uncertainty
6. Construct integrated safety assessment model and perform calculations
7. Perform uncertainty and sensitivity analyses
8. Evaluate performance
9. As needed, direct science and testing program

Figure 2-16 shows the sequential and iterative nature of these nine steps.



Source: Meacham et al. 2011.

Figure 2-16. Safety Assessment Methodology

Details of the general application of each of the nine steps are provided in (Meacham et al. 2011, Section 1.2.2). Brief summaries from that document are provided below, with references to later sections of this report where the specific application to the four geologic disposal options is further described.

Step 1: Define Performance Goals—Performance goals are typically defined up front because they determine the design of the performance assessment analysis and have considerable influence on scenario construction, model development, and research programs. In the case of deep geologic disposal projects, the performance goals are typically regulatory standards and requirements. For early iterations of the safety assessment methodology, assumptions about possible future standards may need to be made. Goals and criteria for the postclosure performance of generic geologic disposal systems are discussed in Sections 2.3.2 and 4.1.

Step 2: Characterize System—A system description includes the characteristics of the waste (e.g., radionuclide inventory, decay chains, half-lives), the facility (e.g., size, thermal loading from emplaced waste, design, properties of engineered barriers), and the site (e.g., geology, hydrology, geochemistry). System information is derived from laboratory and field tests, published literature, natural analogues, and/or expert judgment. In the early stages of a nuclear waste disposal program, the characterization program is broad-based and focused on gaining an adequate understanding of the system, identifying the greatest sources of uncertainty, and identifying the FEPs most likely to affect the system's long-term performance. As knowledge and understanding of the system improve, resources are allocated to characterization activities that most likely will result in reasonable reductions to important sources of uncertainty.

Characteristics of generic repository layouts and concepts, waste types, engineered barriers, and natural barriers are described in Sections 2.2 and 3.

Step 3: Develop Scenarios—The processes and events, or sequences of processes and events, which may be relevant over the timeframe of interest, need to be identified and included in the quantitative safety assessment. This step, generally referred to as scenario analysis, consists of two activities: FEP analysis and scenario development.

FEP analysis involves (1) identifying and classifying the FEPs potentially relevant to the long-term performance of the system of interest, (2) screening the potentially relevant FEPs to identify those FEPs that are important to system performance and must be included in the performance assessment analysis, and (3) documenting the rationale for exclusion of those FEPs not important to performance. An important goal in identifying the FEPs potentially relevant to long-term performance is the demonstration of completeness (i.e., nothing is too insignificant or improbable to be considered as potentially relevant). The demonstration of completeness addresses a key source of uncertainty in a safety assessment.

Section 4.2.2 discusses general considerations for FEP analysis and then describes FEP identification and screening specific to the generic disposal systems considered in this safety case report.

Scenario development involves (1) constructing a set of potential scenarios – combinations of included (or retained) FEPs that representation a possible future state of the system, (2) screening the potential scenarios to identify those scenarios that are important to system performance and must be included in the performance assessment analysis, and (3) documenting the rationale for exclusion of those scenarios not important to performance. All included FEPs must be accounted for in the safety assessment in at least one scenario. A typical approach to scenario development is to create an "expected" or nominal scenario and one or more "disturbed" scenarios.

Section 4.2.3 discusses general considerations for scenario development and then describes scenario screening specific to the generic disposal systems considered in this safety case report.

Step 4: Build Models and Abstractions—The FEPs and scenarios retained after the screening process are represented in the quantitative safety assessment through conceptual models, mathematical models, and computational (numerical) models. Conceptual models describe system behavior for the scenarios of interest (e.g., Darcy flow in porous media). Conceptual models consist of a representation of the included FEPs in each scenario (e.g., groundwater flow), the different laws of physics and chemistry that govern those FEPs (e.g., Darcy's Law), the dimensionality of the model (one-, two-, or three-dimensional model), and other assumptions (e.g., time-dependent or steady state). Mathematical models quantify conceptual models in terms of mathematical expressions ranging from simple representations (e.g., response surfaces, independent linear relationships) to complex representations (e.g., coupled nonlinear partial differential equations). Computational models provide numerical (or analytical) solutions to the mathematical models.

The models consist of sets of hypotheses, assumptions, and simplifications (abstractions) that, together, describe the essential aspects of a system or subsystem of the repository relative to performance. Because the safety assessment methodology deals with future outcomes and includes uncertainty in both descriptions of FEPs and parameter values, an essential element of the safety assessment is to capture uncertainty in probabilistic analyses that represent likely outcomes, based on the best available values of process model parameters and the processes involved.

The conceptual, mathematical, and computational models describing the generic disposal system FEPs and scenarios are discussed in Section 4.3.

Step 5: Quantify Uncertainty—Three major sources of uncertainty must be considered in a safety assessment (Meacham et al. 2011, Section 1.2.1):

- Uncertainty in the future state of the system
- Model uncertainty
- Data and parameter uncertainty

Uncertainty in the future state of the system arises because it may be difficult to forecast exactly how the disposal system will evolve over tens to hundreds of thousands to millions of years. The system's evolution can be affected by natural events (e.g., seismicity) as well as human-induced events (e.g., drilling). This form of uncertainty is referred to as aleatory uncertainty and is also known as stochastic, type A, or irreducible uncertainty. Aleatory uncertainty is typically addressed in a performance assessment model through scenario construction and screening, where each retained scenario represents a possible future state of the disposal system. Scenario probabilities are used to weight the consequences of each scenario according to its probability of occurrence.

Each scenario is represented by a set of conceptual, mathematical, and computational models that are used to estimate of the consequence(s) if the scenario was to occur. Independent of their complexity, models are simplifications of reality; therefore, model development necessarily entails making assumptions. Model uncertainty is introduced due to (1) the accuracy and appropriateness of the assumptions, (2) the completeness, or exhaustiveness of the assumptions, and (3) the complexity of interactions between the assumptions. Model uncertainty may be addressed through the use alternative conceptual, mathematical, or computational models.

Data and parameter uncertainty arises from incomplete knowledge of the present system and the inherent complexity of natural systems. Often, the parameter values are not measured directly but rather are inferred from measured data that require interpretation. Uncertainty in measured data can be introduced from limitations in measuring instruments, inability to fully characterize spatial and other variabilities, or human error. Uncertainty in parameter values comes from the uncertainty in the measured data as well as the uncertainty that may be associated with data interpretation. Many parameters used in the performance assessment models of geologic repositories, such as parameters derived from site data on common rock properties (e.g., porosity and permeability), have natural variability. Data and parameter uncertainty also arises when, for example, the future state of a changeable property—which cannot, obviously, be measured—must be estimated. This form of uncertainty is referred to as epistemic uncertainty and is also known as subjective, state-of-knowledge, type B, or reducible uncertainty. Epistemic uncertainty is typically accounted for by developing distributions of values for important, imprecisely known parameters rather than using single deterministic values. Each distribution describes a range of values within which the true value is believed to fall, with an expected value that corresponds to the best estimate of the true value.

Section 5.2 discusses general considerations for uncertainty quantification of geologic disposal systems.

Step 6: Construct Integrated Safety Assessment Model and Perform Calculations—An integrated safety assessment model is required to estimate the consequence(s) of the retained scenarios and the overall probability-weighted consequence(s) of the disposal system considering all possible scenarios. The integrated safety assessment model is constructed by (1) coupling the sets of FEP and scenario submodels together to calculate overall system performance, and (2) identifying values and/or uncertainty distributions for system model input parameters. Integrated generic safety assessment models for each of the four geologic disposal options are described in Section 4.3.

Uncertainty in the input parameters can be treated using deterministic or probabilistic methods. In a deterministic simulation of a specific scenario, each input parameter is assigned a single value, typically representative of best estimate or baseline conditions. The integrated safety assessment model is then used to calculate a corresponding value(s) for the system performance measure(s) under those baseline conditions. Uncertainty can be treated by varying one or more parameter values from the baseline value and examining the corresponding effect on the system performance. This type of deterministic sensitivity analysis is known as a “one-off” analysis. Another type of deterministic simulation is a bounding analysis, in which parameter values are selected such that the performance of the system is “worst case”. While defining what the worst case is can be a challenge, it is typically easy to defend if all agree that the performance could not be worse than that calculated.

In a probabilistic, or stochastic, simulation of a specific scenario, the parameter values are sampled and propagated through the coupled set of models to generate a distribution of potential outcomes. Parameter uncertainty is propagated into the safety assessment by conducting multiple calculations for each scenario using values sampled from the distributions of possible values (e.g., Monte Carlo simulation). Each individual calculation uses a different set of sampled input values and produces a different value(s) for the system performance measure(s). In a statistical sense, the result of each individual calculation represents a different possible realization of the future overall performance of the system, consistent with the uncertainty in the input parameters. Overall system performance for a specific scenario is then quantified by some measure of the distribution of results from all realizations, such as the mean or median of the system performance measure(s). Uncertainty associated with the probability of occurrence of each scenario is included in the safety assessment by conducting separate analyses for each scenario and then probability weighting the results to estimate an overall system consequence.

As a geologic disposal system program matures, uncertainties in disposal system performance models are typically treated using probabilistic methods. However, for the safety assessment models used to support this initial safety case, deterministic baseline simulations of each of the four geologic disposal options were performed. Deterministic simulation results are described in Section 4.4.1.

Step 7: Perform Uncertainty and Sensitivity Analyses—The results of sensitivity analyses can be analyzed further using statistical techniques, such as step-wise linear regression, to identify and rank the importance of uncertain parameters. Caution must be used in interpreting the results of the sensitivity analysis because the rankings are conditional on the modeling assumptions and parameter distributions, because the processes not included in the models cannot be evaluated, and because the sensitivity of fixed-value parameters cannot be evaluated.

The sensitivity analysis identifies parameters for which reductions in uncertainty will reduce the uncertainty in the estimate of overall system performance. Sensitivity analyses are included in the safety assessment to enable the process to be iterative, providing feedback during research and development activities to efficiently and effectively reduce important sources of uncertainty. The sensitivity analyses also allow the detailed performance assessment models to be simplified for more rapid iterative calculations by “turning off” unimportant functions. Thus, the results of the sensitivity analysis are used to guide programmatic decision-making, which is especially important as the project evolves.

Section 4.4.2 discusses the deterministic one-off sensitivity analyses performed for the four generic disposal systems.

Step 8: Evaluate Performance—Quantitative safety assessment results provide indications of subsystem and overall system performance. When combined with sensitivity analyses, safety assessment results can be used to identify the models and parameters that have the greatest effect on the behavior of the system. Identification of the uncertainties that are most important in preliminary safety assessments can help guide site characterization, repository design, and model development through a directed science and testing program. When the safety assessment models are sufficiently well developed and documented to support regulatory decisions, results can be used to evaluate compliance with applicable long-term requirements. The steps in the safety assessment process are repeated, as needed, until a final decision is reached.

The generic safety assessment model results are evaluated in Sections 4.4 and 5.2.3.

Step 9: Directed Science and Testing Program—Information from the overall performance evaluation and uncertainty and sensitivity analyses serves to identify important parameters and systems for further investigation. This may include identifying systems whose performance can be improved by modifications to the design, or parameters with uncertainties that, if reduced through further site or laboratory investigations, would significantly increase confidence in the overall safety assessment results. The safety assessment process thereby helps inform programmatic decision-making toward the testing and scientific investigations that will most effectively and efficiently improve the accuracy and confidence in safety assessment results and toward design decisions most likely to improve real system performance.

In summary, the iterative application of the safety assessment methodology through the lifetime of a deep geologic disposal project supports a robust and defensible:

- Evaluation of subsystem and total system performance with respect to specific criteria or requirements
- Consideration of expected and disturbed scenarios
- Evaluation of design options/alternatives
- Development and streamlining of the models used to simulate the important FEPs and scenarios
- Determination and representation of significant sources of aleatory, epistemic, and model uncertainty
- Incorporation of information from laboratory and field tests, published literature, natural analogues, and expert judgment
- Prioritization of research and testing needs

2.3.2 Safety Crieria

The basis for evaluating the suitability of a site to proceed from one phase to the next depends on an evaluation against safety criteria. In the U.S., such criteria are promulgated by the U.S. Environmental Protection Agency, the U.S. Nuclear Regulatory Commission, and the U.S. Department of Energy. The safety standards and the implementing regulations governing development of a repository are the fundamental technical requirements that are addressed in a safety case and safety assessment. Safety standards and guidance are also available from international groups and programs.

Isolation of nuclear wastes depends on radionuclide containment, on limiting potential releases, and attenuation of these releases such that potential future radiation exposures are at levels that are in accordance with applicable standards of safety. Disposal of long-lived radioactive waste in repositories or engineered facilities located deep underground in suitable geological formations is being widely investigated worldwide in order to protect humans and the environment both now and in the future. A repository is considered to be safe, from a technical point of view, if it meets the relevant safety standards. Internationally recommended standards or guidance may also be used to evaluate safety.

The U.S. Nuclear Regulatory Commission regulations are moving from addressing uncertainty by ensuring that individual components provided certain degrees of performance, to a risk-informed, probability-based assessment of the performance of the system as a whole. Scientists have recognized for years that the subsystem-based approach did not provide assurance that the repository would perform as well as it was able to (McCartin 2010). A risk-informed, probability-based approach, on the other hand, emphasizes the performance of the system as a whole, while allowing for the demonstration of which components of the repository contribute to performance.

This Generic Safety Case focuses on performance objectives that are internationally acknowledged to be important to the disposal concept at any stage during implementation. For the purposes of this Generic Safety Case, the following performance objectives for disposal systems are adopted (U.S. Department of Energy 2012, modified from International Atomic Energy Agency 2003a) to provide a basis for evaluating performance:

- **Containment**—Provide a high probability of substantially complete containment of short-lived radionuclides for some hundreds or thousands of years, perhaps largely within the engineered barriers of the repository
- **Limited Releases**—Limiting and/or delaying the rate and the consequent concentrations in which radionuclides will be released from the immediate environment in which the waste was emplaced into the surrounding geological environment and eventually transported to the biosphere. This is achieved by a combination of physical and chemical mechanisms which, among other functions, may limit and/or delay the access and flux of groundwater to the wastes and from the repository to the biosphere, and may limit the solubility of radionuclides, or sorb or precipitate them reversibly or permanently onto surfaces in the host geology and the engineered barrier system. In addition, the process of radioactive decay progressively reduces the amounts of radionuclides present in the disposal system (although the amounts of some important radionuclides will increase through in-growth).
- **Dispersion and Dilution**—The flux of long-lived radionuclides through the geological barriers involves three-dimensional dispersion, and may take place in widely different groundwater environments. In some concepts and at some specific proposed repository sites, releases would encounter major aquifers at depth or closer to the surface, or similar large bodies of surface water. This would result in an additional, but secondary, function to limiting releases (i.e. an overall dilution of released radionuclides such that concentrations on initial return to the biosphere are lowered).
- **Defense in Depth**—Ensured by performance of a geological disposal system dependent on multiple barriers having different safety functions

Section 2.3.2.1 provides a historical overview of the types of criteria applied to earlier risk assessments and evaluation of performance for geologic disposal systems. Section 2.3.2.2 presents a description of the types of performance measures that may be used to assess the long-term performance of a generic geologic disposal system.

2.3.2.1 Early Criteria Considered for Evaluating Geologic Disposal

A review of the historical development of criteria and guidance that have been considered or used in siting is presented in this section. At the siting stage, safety assessment capabilities are necessarily limited because their development depends on the focused understanding of the site specific natural and engineered system component behavior needed to develop total system models. As a result, the screening criteria at this stage largely place emphasis on subsystem components of a geologic disposal system and they help identify and focus attention on those system components that potentially could be barriers important to waste isolation. Locating potential sites with features that that tended to be favorable with respect to such criteria likely would also focus on areas that are structurally stable, geologically old, large and deep, with long groundwater travel times, and geochemical conditions that favored chemical stability.

It is unlikely (and unnecessary) that a site be ideal with respect to all aspects of applicable screening guidance or criteria. It is sufficient to establish that an adequate, safe, and acceptable site has been identified where it would be expected that the site and design, in combination, would meet all applicable safety requirements. The likelihood of success in selecting a site for further study could possibly be enhanced for those sites that exhibit appropriate combinations of the site attributes that could favor geologic disposal system performance. It is appropriate to note site screening will ultimately involve tradeoffs between attributes that could favor geologic disposal system performance and those that may present challenges.

The criteria summarized from previous screening guidance presented here emphasize the geotechnical character of a potential site; comprehensive screening guidance would also typically include attributes of the engineered components of the system, as well as social and environmental considerations. Because a principal purpose of this Generic Safety Case is to introduce the initial generic postclosure safety assessments for the four geologic disposal options, it is reasonable to address the previous criteria most relevant to safety assessments for these four disposal options. Future site selection efforts will likely need to consider a broader set of guidance factors.

The National Research Council developed an early, comprehensive set of *Geological Criteria for Repositories for High-Level Radioactive Waste* (National Academy of Sciences 1978). The criteria set included geological criteria and geo-economic criteria. There were three geo-economic criteria, and they were exclusionary:

- No area with a present or past record of resource extraction, other than for bulk materials won by surface quarrying, should be considered as a geological site for radioactive waste
- No area should be considered as a potential site for a repository unless sufficient geological information is at hand to provide a basis for a reasonable analysis of resource potential
- No area adjacent to an actual or potential major dam site should be considered as a potential site for repository

The geological criteria addressed geometrical and dimensional aspects of a potential facility, long-term stability, hydrology, and geochemistry. The three geometrical and dimensional criteria were:

- The repository should be at depth sufficient to separate the repository from any surficial process or event that might cause a breach of the repository
- The size and shape of the specific body of rock in which a repository is to be constructed should be adequate to allow room for both the repository and also a sufficiently large buffer zone around the repository
- The availability of information on the geometry and physical, chemical, and mineralogical properties of the prospective host rock body and the associated rocks in advance of development of the site

The geometrical and dimensional criteria were augmented by five long-term stability criteria, two of which were exclusionary in nature:

- The repository should lie within a structurally stable geological block and not near a tectonic boundary
- Faults along which rupture could occur must be avoided
- Areas with abnormally high geothermal gradients or with evidence of relatively recent volcanic activity are possible candidates for future volcanic events and should be avoided
- The mechanical geophysical properties and the state of stress in the repository host rock should be such as to assure the stability of the repository during its operation

- Backfilling and sealing each segment of the mine cavity should be accomplished as soon as the waste is in place and the final checking and proving of that part of the facility has been completed

While not strictly a long-term stability criterion, National Research Council recommendations included an admonition that to keep the design and operation of repository simple, and to reduce the number of variables in the total system, unchopped and unreprocessed fuel elements should not be placed in retrievable storage in a geologic repository designed for the permanent disposal of radioactive waste.

The three hydrological and four geochemical criteria recommended by the National Research Council were

- Hydrologic analysis of the perturbed geologic system involving a repository must determine that fluid transport will not move hazardous material to the biosphere in amounts and rates above prescribed limits
- Because the vertical shafts of an underground repository may be the most probable route for the hydrological transport of radionuclides to the active biosphere, a geological system should be selected that can be satisfactorily plugged and sealed when the repository is closed, and suitably monitored to ensure that the behavior of the overall hydrogeological system will continue to function satisfactorily after closure
- The geological record of previous hydrogeological conditions, or the paleohydrogeological record, should be such that predictions can be made that are favorable for long-term hydrological isolation of the repository site in a perturbed geological environment
- Radioactive heat and radiation should not reach levels high enough to produce physical and chemical reactions in the repository rock that would compromise the geological containment
- The interaction of water, repository rock, and the waste material should be controlled in such a way as to minimize the rate of dissolution of the waste form
- Water in the repository, if present, should not react chemically or physically with the repository rock to increase its permeability which would compromise geological containment
- The properties of the geochemical system of the radionuclides, the repository rock, and its associated water should be such as to restrict or prevent the mobility of the radionuclides and to delay or prevent their migration to the active biosphere

A subgroup of the Earth Science Technical Plan Working Group of the U.S. Department of Energy and the U.S. Geological Survey prepared a plan in 1980 for identification and geological characterization of sites for mined radioactive waste repositories (U.S. Geological Survey 1980). The plan involved a process that was designed for screening to successively smaller land units. The screening process used four principal steps for each stage: (1) characterization of land units; (2) delineation of subunits; (3) identification (rating) of subunits; and (4) the selection of subunits for more intensive characterization than the next stage. It envisioned screening at the national level, province level, region level, area level and potential site level.

The Earth Science Technical Plan Working Group report included the following screening criteria:

- **Repository Host Rock**—The area should be underlain at least in part by a system of rocks containing one or more suitable host rock units. The following factors should be considered:
  - *Mineability*—It should be possible to mine the facility using available mining methods and technology.
  - *Thermal Conductivity*—The thermal conductivity of the rock unit should be high enough to accommodate thermal stresses imposed by the particular waste form without causing serious increase in hydraulic conductivity or detrimental alteration of the waste form.

- *Fractures*—The rock unit should have a minimum of natural and induced hydraulically conductive fractures.
- *Hydraulic Conductivity*—The rock unit should have a low hydraulic conductivity.
- *Dimensions and Geometry*—The rock unit should be a sufficiently thick and extensive area to accommodate the facility.
- *Depth*—The repository host rock should occur at sufficient depth to minimize the possibility of exposure through geomorphic and or tectonic processes.
- *Homogeneity*—The rock unit should be sufficiently homogeneous to make it possible to predict its essential long-term physical properties in advance of mining and development.
- *Sorption Capacity*—The rock unit should have a high radionuclide sorption capacity to enhance radionuclide residence time in the flow system.
- *Geochemical Properties*—The geochemical properties of the rock unit and contained water should not result of chemical reactions with the wastes which would facilitate the transport of radionuclides from repository sites.

• **Groundwater Flow System**—The essential attribute of the flow system is that it provides long residence time before the water enters the biosphere.

- *Travel Time*—The rocks in the system between the repository site and the discharge area should have a low hydraulic conductivity, along the flow path, and small hydraulic gradients to provide long residence time.
- *Flow Direction*—Groundwater in a substantial part of the area should have strong downward or lateral component of flow. There should be no upward flow, particularly if the area contains numerous oil, gas, or other exploratory boreholes, or a high potential for such holes being drilled in the future.
- *Uniformity*—The hydraulic characteristics of the system should be sufficiently uniform to permit the spatial extrapolation of these characteristics to the nearest discharge area. In general, it is preferable that the rocks along the flow path should have granular rather than fracture porosity.
- *Sorption*—Rocks with a high sorptive capacity should dominate along the groundwater flow paths.
- *Water Quality*—To minimize the possibility of future intrusion of the repository by water well drilling, the potential host rock unit should be underlain by, and at least immediately be overlain by, nonpotable water. A potable aquifer system near the surface would minimize the incentive to drill deeper in search of water.

• **Tectonic Conditions**—Certain geologic processes resulting from tectonic activity could disrupt the repository environment and facilitate the mobilization of waste radionuclides in groundwater, or possibly even directly expose the waste. The criteria should consider the following areal factors in terms of their effects on potential repository environments:

- Known active faults
- High seismic intensity
- Recent volcanic activity
- Persistent uplift

• **Mineral Resources**—The intent is (1) to avoid mineralized zones at depths greater than the potential host rocks to minimize the possibility of radionuclides escaping from the repository through pre-existing boreholes which could not be sealed satisfactorily; and (2) to minimize the potential of penetrating the repository in the future by holes drilled for mineral exploration or development. In this sense aquifers containing water of potable quality are considered a mineral resource.

- **General Considerations**—It will be difficult to develop a universally acceptable set of criteria involving geologic processes, many of which are imperfectly understood. The point cannot be stressed strongly enough that the use of a generally applicable set of criteria will require a great deal of insight and perception. In addition to the set of criteria to be developed for screening regions, at least one more set will have to be developed to consider specific, unique features of areas and differing rock types, and to judge the relative merits of potential sites. For candidate sites, however, additional sets of criteria will have to be quite detailed and specific about local geologic characteristics; thus, suggestions for their development cannot be made at this time.

2.3.2.2 Metrics To Assess the Long-Term Performance of a Generic Geologic Disposal System

The focus of the historical criteria in Section 2.3.2.1 emphasized screening sites for suitability for a geologic disposal facility. A summary of those screening criteria are presented in Table 2-5.

Table 2-5. Historical Generic Screening Criteria

Criteria Source	Category	Examples
National Academy of Sciences (1978)	Geo-Economic	Resource potential; avoid dams
	Geologic - Geometric	Adequate depth and size
	Geologic - Stability	Stable area; stable openings; avoid faults
	Hydrologic	Long-term hydrologic isolation; sealing
	Geochemical	Limit geochemical reactions
Earth Science Technical Plan Working Group (U.S. Geological Survey 1980)	Repository Host Rock	Mineability; minimum fractures; adequate depth
	Groundwater Flow System	Long residence time; nonpotable water
	Tectonic Conditions	Effects of faulting, seismicity, volcanism, uplift
	Mineral Resources	Avoid mineralized zones
	General Considerations	Need for site-specific criteria

A recent study (Hansen et al. 2011) of potentially suitable formations for siting a geologic disposal facility in the U.S. noted that suitable host rocks for deep geologic repositories would typically exhibit certain basic favorable physical characteristics, along with key geologic and hydrologic attributes. Key physical characteristics that could be considered favorable to siting include:

- **Depth**—The disposal horizon should be determined based on site-specific conditions. Geologic isolation is attained by ensuring significant separation between the repository and the biosphere, which would provide extensive zones for robust seal systems. Rock strength characteristics would also determine a practical and functional mining depth. For deep borehole concepts, proposed disposal zone depths are 2-5 km.
- **Thickness**—Maximal thickness of the isolation medium is desired to ensure radionuclide migration does not exceed performance metrics or boundaries. Various “minimal” thicknesses have been put forward, generally of the order of 100 m. The authors concluded that the thickness of the formation is less important than its uniformity and structure. However, thickness could still be considered an important characteristic of a primary barrier.

- **Uniformity and Structure**—The potential repository interval and surrounding rock should be reasonably homogeneous both vertically and horizontally. The related benefits are simpler and more transparent characterization and performance assessments and safer repository mining and operations.
- **Seismicity**—Seismically quiescent regions favor repository design, operations, and long-term performance.

Key geologic and hydrologic attributes of the host rock include:

- **Hydrogeology**—Low hydraulic conductivity (~10^{-12} m/s or lower).
- **Self-sealing**—Rocks with plastic deformation characteristics reestablish a diffusion-dominated transport system.
- **Hydrogeochemistry**—Reducing chemical conditions minimize corrosion of engineered barriers and waste forms, reduce most radionuclide solubilities, and improve sorption. Oxidizing environments are also possible but would require very low hydraulic flux as found in desert environments.

The likelihood of success in selecting a site for further study is enhanced for those sites that exhibit combinations of the potentially favorable attributes of the above siting criteria. The summaries of historical siting efforts presented in Appendix A indicate that numerous areas and sites can be found that potentially meet such site screening criteria. It is reasonable to expect that future screening criteria will not differ substantively from those used in the past.

However, as noted in Section 1.1, it is not the intent of this Generic Safety Case to identify and screen sites for their suitability for a geologic disposal facility. Rather, the Generic Safety Case seeks to provide confidence that used nuclear fuel and high-level radioactive waste can be disposed of safely in the U.S. in mined disposal facilities in salt, clay, and granite formations, in deep borehole disposal in crystalline rocks.

To support a conclusion that sufficient areas exist in the U.S., the most meaningful approach seems to be to demonstrate that the results of the generic safety assessments presented in this Generic Safety Case are not inconsistent with performance metrics previously used. The values in the safety assessment results are presented simply as an indication of the potential for future, site-specific estimates of performance to be acceptable. An assumption exists that performance metrics used in future regulatory criteria are not expected to differ significantly from those used in the past. Based on recent responses to questions about this subject (McCartin 2012, p. 154), it appears likely that the assumption is appropriate. The dose standards in U.S. regulations for geologic disposal are summarized in Table 2-6.

Table 2-6. Dose Standards in U.S. Regulations

Source	Description	Requirement
40 CFR Part 191	§191.13 Containment Requirements	Likelihood of less than one chance in 10 of exceeding the quantities and a likelihood of less than one chance in 1,000 of exceeding ten times the quantities of Table 1 [of the regulation][a]
40 CFR Part 191	§191.15 Individual Protection Requirements.	Annual committed effective all potential pathways dose, received through to any member of the public in the accessible environment, not to exceed 15 mrem (150 μSv)[a]
40 CFR Part 197	§197.20 What Standard Must DOE Meet?	Annual committed effective dose equivalent to reasonably maximally exposed individual no more than (1) 150 μSv (15 mrem) for 10,000 years following disposal; and (2) 1 mSv (100 mrem) after 10,000 years[b]

NOTE: [a]For 10,000 years.
[b]Within the 1,000,000-year period of geologic stability.

The U.S. regulations also include standards covering groundwater protection and inadvertent human intrusion. Groundwater protection is site specific, and human intrusion scenarios are stylized and site specific.

Standards considered by other programs vary; there is, however, an international consensus standard developed by the International Atomic Energy Agency (2011). The Safety Objective of that standard for postclosure performance is the following: a disposal facility (considered as a single source) is so designed that the calculated dose or risk to the representative person who might be exposed in the future as a result of possible natural processes affecting the disposal facility does not exceed a dose constraint of 0.3 mSv (30 mrem) in a year or a risk constraint of the order of 10^{-5} yr^{-1}. Further, the International Commission on Radiological Protection has a draft position on radiological protection in geological disposal of long-lived solid radioactive waste (Weiss et al. 2011). They recommend an annual dose constraint of 0.3 mSv (30 mrem) in a year to be used for the sake of comparison of options rather than as means of assessing health detriment. This criterion is not dissimilar to the existing U.S. and international criteria. The Nuclear Energy Agency also recommends consideration of flux-related indicators of safety that provide information on the transport of radionuclides between the components of the repository and their release to a receptor (Organisation for Economic Co-operation and Development 2012).

The safety assessment results presented in Section 4.4 examine potential doses to receptors over times as long as 10,000,000 years. Such a time period is far longer than any quantitative requirement applied under current U.S. regulations, or, for that matter, regulations in any other country. This was done principally to gain insight into very long-term behavior of radionuclides. The results included in Section 4.4 should only be viewed as a tool used to gain insight into the potential time of peak doses.

3 ASSESSMENT BASIS FOR GENERIC DEEP GEOLOGIC DISPOSAL SYSTEMS

3.1 Overview

As seen in Figure 1-1, the role of the assessment basis in a safety case is typically to present the quantitative information addressing site selection, site characterization, and repository design. Within the safety assessment methodology in Section 2.3.1, the assessment basis is captured in Step 2: Characterize system (waste, facility, site). Because only generic disposal options are considered for this safety case, the assessment basis relies less on site-specific quantitative information and more on assumptions and generic descriptions.

The characteristics of the potential inventory of used nuclear fuel and high-level radioactive waste that could be included in a future U.S. deep geologic disposal facility are reviewed (Section 3.2). The assessment basis also discusses multiple barriers of a generic geologic disposal facility system, including factors that affect the ability of the barriers to limit or delay radionuclide transport. Features of the natural barrier system were discussed in the siting strategy in Section 2.2.2.1. For the assessment basis, the relevant processes acting on a natural barrier system are summarized, and then the impact of various processes on the natural barriers of each of the four disposal options is described (Section 3.3). Components and features of the engineered barrier system, as well as their functions, are described in the design strategy in Section 2.2.3.1. The assessment basis provides additional information on the potential waste types and waste packaging, followed by a summary of the processes that act on the engineered barrier system (Section 3.4). The potential impact of external events such as climate changes, seismic events, igneous activity, and human intrusion on the performance of both the natural and engineered barrier systems is discussed in Section 3.5.

Numerous safety assessments pertinent to the four disposal options have already been conducted by the U.S. and other countries around the world. Based on extensive laboratory and field research, these assessments provide insight into potential repository behavior and the complementary role of engineered and natural barriers. Examples of these safety assessments are provided in Appendix C and summarized in Section 3.6. They were conducted by the U.S. and other countries for a range of applications, from generic scoping studies, to safety assessments performed to support safety case development, to safety assessment calculations included in license applications. The results of generic postclosure safety assessments presented in Section 4 are in general agreement with the results of these previous safety assessments.

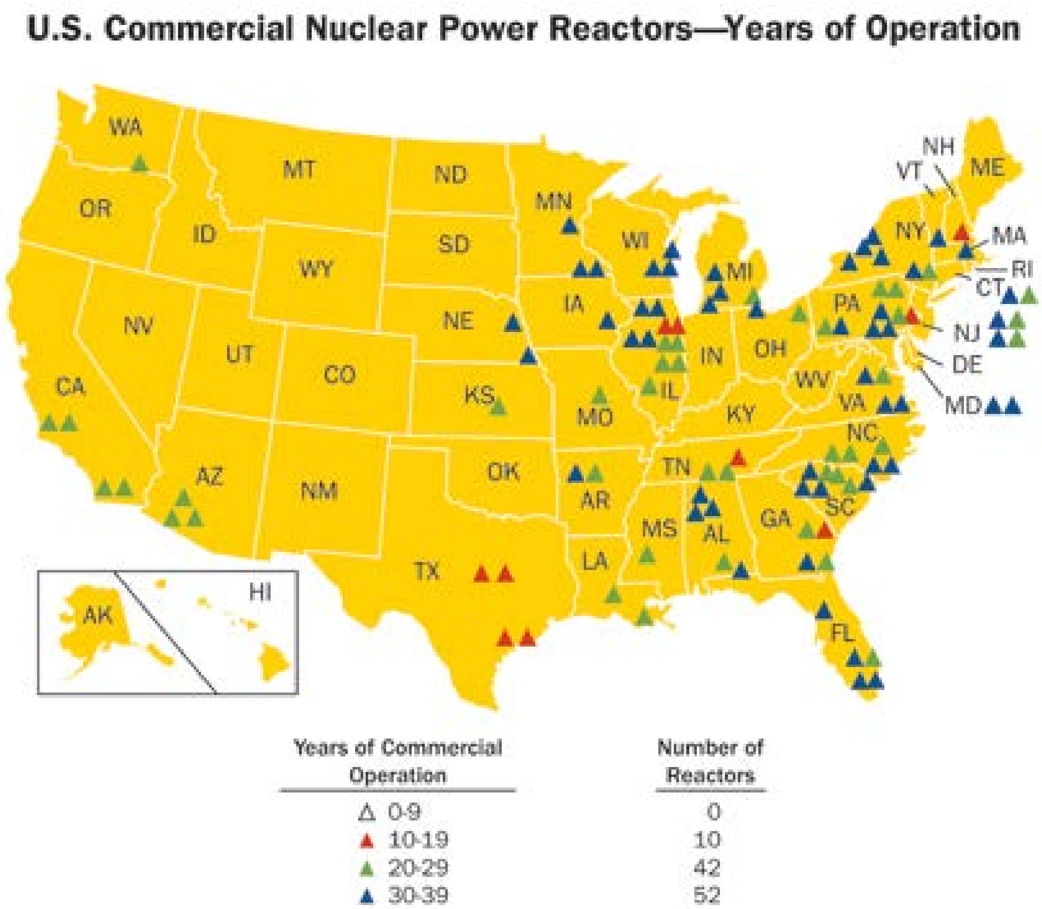
3.2 Potential Inventory Characteristics

The generic safety assessments prepared for this Generic Safety Case are based on an inventory of used nuclear fuel and high-level radioactive waste that currently exists, or can reasonably be projected based on information available at the time of preparation of this document. Designing a facility and assessing safety requires knowledge of the inventory and characteristics of the wastes to be emplaced.

Given the uncertainty about future fuel cycles that could be deployed in the U.S., this Generic Safety Case considers a wide range of potential future waste streams, waste forms, and radionuclide inventories. This section summarizes information about the amounts and characteristics of the following inventory sources: (1) commercial used nuclear fuel from the once-through light water reactor fuel cycle, (2) U.S. Department of Energy used nuclear fuel and high-level radioactive waste, as well as Naval used nuclear fuel, and (3) waste from reprocessing commercial light water reactor used nuclear fuel. Unless otherwise noted, the material in this section is taken from, or follows Carter et al. (2011c) or Carter et al. (2011b); the second reference is a synopsis of the first.

3.2.1 Commercial Used Nuclear Fuel from the Once-Through Fuel Cycle

Commercial nuclear power plants have operated in the U.S. since 1957, when the Shippingport reactor began to generate power for a commercial market. There are currently 104 operating nuclear power plants at 65 sites. There are decommissioned facilities with storage of used nuclear fuel and/or high-level radioactive waste at 9 additional sites. The U.S. has been committed to a once-through fuel cycle with disposal envisioned in a geologic repository since the late 1970s. Used nuclear fuel from these operating plants is currently stored on site in pools or dry storage casks, awaiting removal by the federal government. In addition, used nuclear fuel from 14 shutdown reactors is currently stored on the reactor sites. The General Electric facility at Morris, Illinois is currently the only licensed used nuclear fuel storage facility in operation that is not collocated with an active or former reactor site. Figure 3-1 shows the locations of these reactor sites, and indicates how many years they have been in operation (U.S. Nuclear Regulatory Commission 2011b).



Source: U.S. Nuclear Regulatory Commission 2011b.

Figure 3-1. Operating Commercial Nuclear Power Reactors in the United States

Current Inventory of Commercial Used Nuclear Fuel—The U.S. Department of Energy developed a database of the amounts of used nuclear fuel to be disposed. The inventory was described in U.S. Department of Energy (2002b, 2007). This includes information collected for used nuclear fuel discharges from 1968 through 2002 on a per assembly basis, and on a batch basis for fuel discharges from 1968 through April 2005.

The specific used nuclear fuel data available from these sources are

- Reactor type (pressurized water reactor or boiling water reactor)
- Number of assemblies
- Burn-up by assembly or batch
- Date of discharge
- Initial uranium loading

To develop an inventory estimate through 2010, Carter et al. (2011c) used fuel discharge predictions developed for the Nuclear Energy Institute in 2005 to estimate the number of assemblies and metric tons uranium (MTU). To estimate the average enrichment and burn-up through 2010, projections made by utilities were used. These projections identified a burn-up increase of 2.38% per year for boiling water reactor fuel and 1.11% per year for pressurized water reactor fuel through 2010; the enrichment increases at the same rate as the burn-up. Comparison of the projections made in 1998 to actual data collected through 2004 show very good agreement:

- Pressurized Water Reactor - actual 46,950 MWd/MTU vs. projected 46,922 MWd/MTU;
- Boiling Water Reactor - actual 43,447 MWd/MTU vs. 42,787 projected MWd/MTU

Table 3-1 provides an estimate of the commercial used nuclear fuel discharged through 2010.

Table 3-1. Estimated Commercial Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) Used Nuclear Fuel Discharged through 2010

Total Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up (MWd/MTU)		Average Age (yr)		Total Radioactivity (Billion Ci)	
PWR	BWR	Total	PWR	BWR	Total	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300	14.9	15.4	16	7

NOTE: The number of assemblies has been rounded to the nearest 200. The estimated fuel discharged has been rounded to the nearest 1,000 MTU. The burn-up has been rounded up to the next 100 MWd/MTU.

Characteristics of the Current Inventory of Commercial Used Nuclear Fuel—The current inventory has an average burn-up of approximately 39.6 GWd/MT for pressurized water reactors and 33.3 GWd/MT for boiling water reactors. Nearly 100% of the fuel currently being discharged exceeds the “high burn-up” threshold of 45 GWd/MT defined by the U.S. Nuclear Regulatory Commission. The maximum burn-up from the current reactor fleet is nearing 60 GWd/MT which is limited by both the 5% ^{235}U licensing basis for the current enrichment and fuel fabrication plants, and the reactor licensing basis to 62.5 GWd/MT.

Figure 3-2 illustrates the decay heat projections for 60 GWd/MT pressurized water reactor fuel at maximum burn-up. The figure provides the total decay heat and isotopic groups with similar isotopic parameters that make up the total. The total decay heat for 1 year cooled fuel from the average burn-up of 40 GWd/MT is 10,500 W/MT or somewhat less than the 14,000 W/MT from the maximum burn-up (60 GWd/MT) fuel.

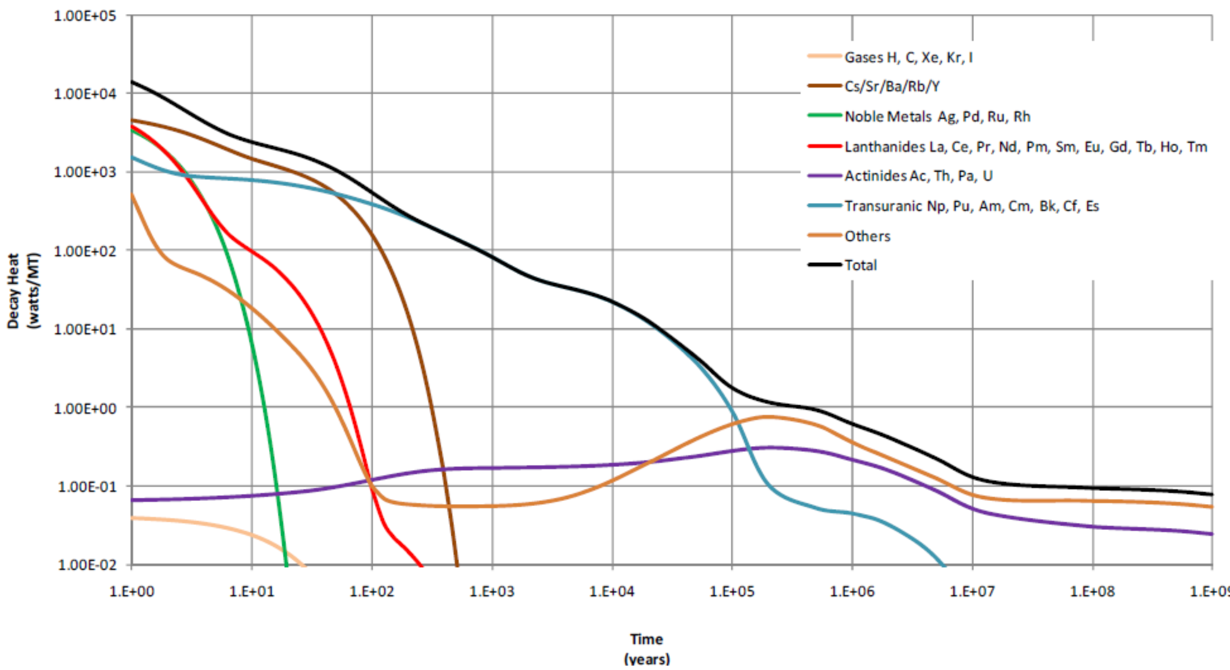


Figure 3-2. Used Nuclear Fuel Decay Heat for Pressurized Water Reactor Fuel at 60 GWd/MT

Projected Inventory of Commercial Used Nuclear Fuel—Four scenarios were used to project the potential inventory of commercial light water reactor used nuclear fuel in the future. The four scenarios were selected from those previously evaluated by the U.S. Department of Energy in the Global Nuclear Energy Partnership Draft Programmatic Environmental Impact Statement, with slight modification for the operational date of the first new reactors (2020 instead of 2015) and the end of the construction period (2060 instead of 2060 to 2070). The scenarios were selected to provide a wide range of light water reactor fuel inventory for use in future analysis; at this time they can be considered reference cases and future research could change the range.

- Scenario 1 assumes no replacement of existing nuclear generation reactors.
- Scenario 2 assumes the amount of current nuclear generation is maintained at the current levels (100 GWe/yr) with new reactors replacing the existing reactors as the existing reactors are decommissioned.
- Scenario 3 assumes the amount of nuclear generation will increase to 200 GWe/yr from 2020 to 2060, and remain at 200 GWe/yr until the end of the century.
- Scenario 4 assumes the amount of nuclear generation will increase to 400 GWe/yr from 2020 to 2060, and remain at 400 GWe/yr until the end of the century.

Figure 3-3 projects the mass (MTU) cumulatively to the end of the century. The inventory can range from about 140,000 MTU assuming no replacement reactors are constructed, to nearly 700,000 MTU assuming nuclear power generation quadruples between 2020 and 2060. The enrichment and burn-up in these scenarios is expected to continue to increase to the current United States enrichment plant limit of 5% ^{235}U, which corresponds to a maximum burn-up of slightly less than 60 GWd/MT.

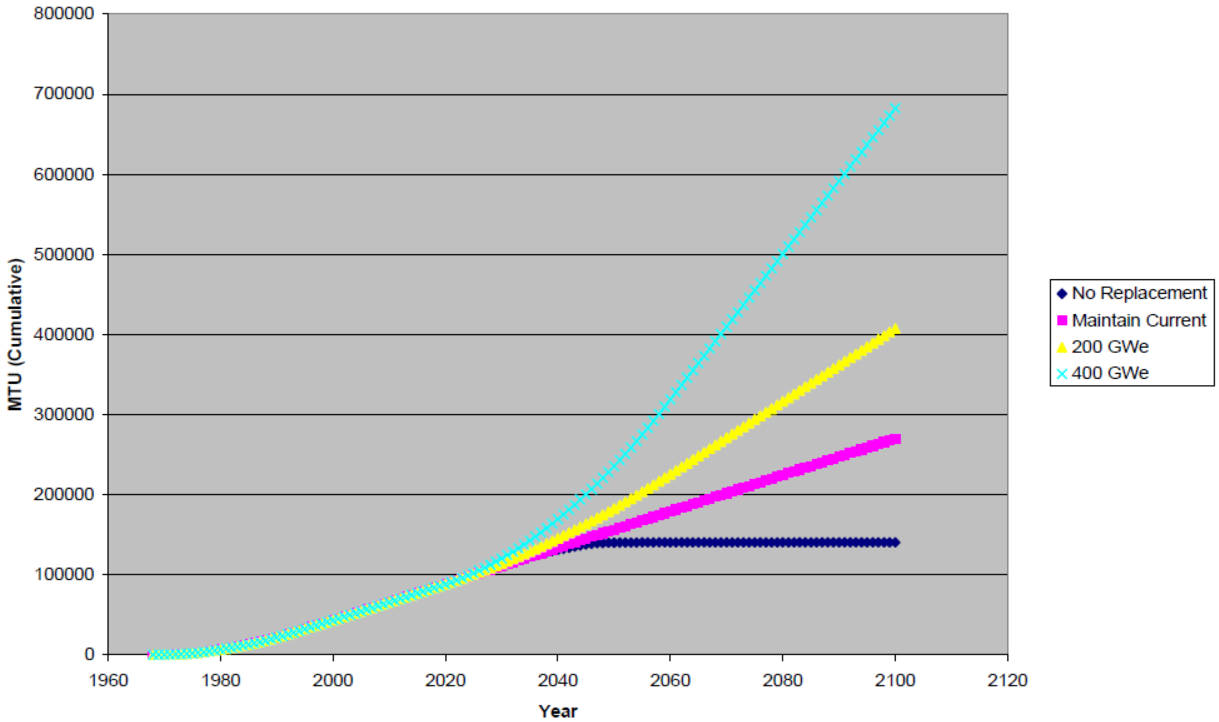


Figure 3-3. Cumulative Used Nuclear Fuel for the Four Cases:
No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr

Maintaining the current nuclear generation rate of about 100 GWe/yr with new reactors replacing the existing reactors as the existing reactors are decommissioned (Scenario 2 above), and assuming the same ratio of pressurized water reactors to boiling water reactors as currently exists results in the creation of the number of assemblies shown in Table 3-2.

Table 3-2. Number of Assemblies, Total Initial Uranium, Average Enrichment, and Average Burn-up, by Year, for Scenario 2

Year	Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burn-up (MWd/MTU)[c]	
	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2010	97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300
2020	131,000	173,000	304,200	57,000	30,900	88,000	4.04	3.57	43,100	39,000
2040	198,000	261,600	460,200	86,600	46,800	133,400	4.36	4.05	46,900	44,900
2060	266,000	350,000	616,000	116,000	63,000	179,000	4.52	4.29	48,800	47,800
2080	333,000	439,000	772,000	146,000	79,000	224,000	4.62	4.43	49,900	49,500
2100	401,000	527,000	928,000	175,000	95,000	270,000	4.68	4.53	50,600	50,600

NOTE: Assumes maintaining current nuclear generation rate of about 100 GWe/yr with new reactors replacing the existing reactors as the existing reactors are decommissioned.

[a]The estimated fuel discharged has been estimated to the nearest 100 MTU prior to 2050 and to the nearest 1,000 thereafter.

[b]The estimated number of assemblies has been estimated to the nearest 200 prior to 2050 and nearest 1,000 thereafter.

[c]The burn-up has been estimated to the next 100 MWd/MT.

3.2.2 U.S. Department of Energy Used Nuclear Fuel and High-Level Radioactive Waste

Since the inception of nuclear reactors, the U.S. Department of Energy and its predecessor agencies operated or sponsored a variety of research, test, training, and experimental reactors both domestically and overseas. The Naval Nuclear Propulsion Program has generated used nuclear fuel from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored and land-based training ship reactor plants, early development of commercial nuclear power, and conduct of irradiation test programs.

Aqueous reprocessing of U.S. Department of Energy used nuclear fuel has occurred at the Hanford Site, the Idaho National Laboratory, and the Savannah River Site. The Savannah River Site is actively vitrifying legacy high-level radioactive waste into borosilicate glass and the Hanford Site has a similar facility under construction. The Idaho National Laboratory has treated high-level radioactive waste in a process known as calcining, and is pursuing the use of electro-chemical processing to treat 60 metric tons heavy metal (MTHM) of sodium-bonded used nuclear fuel. The U.S. Department of Energy is also responsible for clean-up of the commercial used nuclear fuel reprocessing site at West Valley, New York. The wastes requiring disposal from these U.S. Department of Energy activities are fairly well understood and documented.

3.2.2.1 U.S. Department of Energy Used Nuclear Fuel

U.S. Department of Energy used nuclear fuel is generated primarily by Department-owned production reactors, demonstration commercial power reactors, and domestic and foreign research and training reactors. The category of U.S. Department of Energy used nuclear fuel includes some commercial used nuclear fuel that is not in the possession of U.S. Nuclear Regulatory Commission licensed commercial utilities, such as fuel from the Shippingport, Peach Bottom, Three Mile Island, and Fort St. Vrain plants, which is stored at U.S. Department of Energy facilities.

Current Inventory of U.S. Department of Energy Used Nuclear Fuel—U.S. Department of Energy used nuclear fuel comes from a wide range of reactor types, such as light- and heavy-water-moderated reactors, graphite moderated reactors, and unmoderated (fast) reactors, with various cladding materials and enrichments The fuel compositions typically vary from depleted uranium to over 93% enriched ^{235}U. Many of the reactors, now decommissioned, had unique design features, resulting in a diverse spectrum of reactor and fuel designs.

Although U.S. Department of Energy used nuclear fuel is stored throughout the U.S. at numerous facilities, a decision was made in 1995 to consolidate U.S. Department of Energy used nuclear fuel at three existing U.S. Department of Energy sites; the Hanford Site in Washington, the Idaho National Laboratory in Idaho, and the Savannah River Site in South Carolina. The majority of the U.S. Department of Energy used nuclear fuels are currently stored at these three sites. The storage configurations vary for each of the sites and include both dry and wet storage.

A large portion of the U.S. Department of Energy used nuclear fuel (about 2,100 MTHM of the total 2,500 MTHM) is contained in about 400 sealed canisters, almost all of which is N Reactor fuel in dry storage at the Hanford site. The remaining 400 MT of U.S. Department of Energy used nuclear fuel will be placed in an estimated additional 2,500 - 5,000 canisters.

Characteristics of U.S. Department of Energy Used Nuclear Fuel—Conservative used nuclear fuel source-term estimates were developed from process knowledge and the best available information regarding fuel fabrication, operations, and storage. The U.S. Department of Energy used nuclear fuel characterization process relies on pre-calculated lookup tables that provide radionuclide inventories for selected representative used nuclear fuel at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels. Pre-calculated radionuclide inventories are extracted from the appropriate template at the desired decay period and then scaled to account for differences in fuel mass and burn-up for individual items in the inventory. Table 3-3 lists the projected total radionuclide inventory activity of the U.S. Department of Energy used nuclear fuel for the nominal and bounding cases as of 2010. The nominal case is the expected or average inventory. The bounding case represents the highest burn-up assembly or accounts for uncertainties if fuel burn-up is not known.

Table 3-3. U.S. Department of Energy Used Nuclear Fuel:
Total Radionuclide Inventory Curie (Ci) Activity Range as of 2010

Nominal Fuel Inventory (Ci)	Bounding Fuel Inventory (Ci)
1.9×10^8	3.48×10^8

The amount of fuel in each fuel group type, the enrichment, the cladding composition, the fuel matrix and composition, and geometric properties of the fuel group type have been tabulated in detail (Carter et al. 2011c, Appendix D).

3.2.2.2 Naval Used Nuclear Fuel

The Naval Nuclear Propulsion Program generated used nuclear fuel from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored and land-based training ship reactor plants, early development of commercial nuclear power, and the conduct of irradiation test programs. The source of naval used nuclear fuel information is U.S. Department of Energy (2008b).

Current Inventory of Naval Used Nuclear Fuel—Naval used nuclear fuel consists of solid metal and metallic components that are nonflammable, highly corrosion-resistant, and neither pyrophoric, explosive, combustible, chemically reactive, nor subject to gas generation by chemical reaction or off gassing. Approximately 27 MTHM of Naval used nuclear fuel currently exists with a projected inventory of less than 65 MTHM in 2035.

Used nuclear fuel from the Naval Nuclear Propulsion Program is temporarily stored at the Idaho National Laboratory. The existing Naval used nuclear fuel canister designs were developed to fit within the proposed waste package design presented in U.S. Department of Energy (2008b). Approximately 400 naval used nuclear fuel canisters are currently planned to be packaged and temporarily stored pending shipment. The Naval Nuclear Propulsion Program is responsible for preparing and loading naval used nuclear fuel canisters and began canister loading operations in 2002. As of February 2010, 27 naval used nuclear fuel canisters have been loaded and are being temporarily stored at Idaho National Laboratory.

Characteristics of Naval Used Nuclear Fuel—The radionuclide inventory activity by isotope for a representative Naval used nuclear fuel canister five years out of reactor has been tabulated in detail (Carter et al. 2011c, Appendix E).

Table 3-4 provides the total estimated radionuclide inventory for a representative Naval used nuclear fuel canister. A period of five years after reactor shutdown was selected for use in the repository source term analyses.

Table 3-4. Naval Used Nuclear Fuel: Total Radionuclide Inventory Curie (ci) per Canister (5 years after discharge)

Inventory per canister (Ci)	Total Inventory (Ci) (400 canisters)
1.45×10^{6}	~ 580×10^{6}

3.2.2.3 U.S. Department of Energy High-Level Radioactive Waste

High-level radioactive waste is the highly radioactive material resulting from the reprocessing of used nuclear fuel. Following aqueous reprocessing, U.S. Department of Energy high-level radioactive waste is in a liquid form and is stored initially in underground metal storage tanks. Long-term storage of high-level radioactive waste requires stabilization of the wastes in a form that will neither react nor degrade for an extended period of time. The Hanford and Savannah River Sites continue to store the high-level radioactive waste generated at those sites in the liquid form while it awaits final treatment. At the Idaho National Laboratory, the liquid waste was solidified for interim storage by calcination.

High-level radioactive waste glass is formulated to keep waste radionuclides well mixed and immobile in a matrix that is resistant to degradation. Liquid waste and sludges from aqueous reprocessing are neutralized by addition of sodium hydroxide, processed for removal of aluminum, and dried to a calcine form. A mixture of solid materials (frit) is added, containing silicon, boron, sodium, and other elements in minor quantities. Borosilicate glass is obtained by adding boron to achieve at least 5% boron oxide (B2O3) by weight (Lutze 2006). The mixture is fused at a vitrification temperature of 1150°C, and then the molten glass is poured directly into canisters typically made from stainless steel, and solidified (Carter and Luptak 2010).

Current Inventory of U.S. Department of Energy High-Level Radioactive Waste—The Idaho National Laboratory reprocessed used nuclear fuel from Naval propulsion reactors, test reactors, and research reactors to recover uranium and in so doing, generated approximately 30,000 m^3 of liquid high-level radioactive waste. Between 1960 and 1997, the Idaho National Laboratory converted all of its liquid high-level radioactive waste into about 4,400 m^3 of a calcine solid waste form.

The Savannah River Site has reprocessed defense reactor used nuclear fuel and nuclear targets to recover valuable isotopes since 1954, producing more than 530,000 m^3 of liquid high-level radioactive waste while doing so. Through evaporation and vitrification of the waste, the Savannah River Site has reduced this inventory to the current level of about 136,000 m^3 of liquid high-level radioactive waste. The Savannah River Site began vitrifying liquid high-level radioactive waste in 1996 and through December 2010 produced 3,059 2 ft × 10 ft high-level radioactive waste canisters.

The Hanford Site reprocessed approximately 100,000 MT of defense reactor used nuclear fuel between 1944 and 1988 and in so doing generated approximately 2,000,000 m^3 of liquid high-level radioactive waste. After years of storage, the volume is about 220,000 m^3 of liquid high-level radioactive waste. Construction of a vitrification facility to recover the plutonium, uranium, and other elements for defense and other federal programs is currently underway with startup scheduled for 2019.

A commercial fuel reprocessing plant located at West Valley, New York operated from 1966 through 1972 and reprocessed approximately 640 MT of used nuclear fuel to recover the unused uranium. During operations about 2,500 m^3 of liquid high-level radioactive waste was generated. The liquid waste was vitrified between 1996 and 2001 producing 275 high-level radioactive waste canisters that are stored at West Valley (U.S. Department of Energy 1996b; Palmer et al. 2004). Table 3-5 summarizes the current high-level radioactive waste inventory.

Table 3-5. Current and Projected Inventory of High-Level Radioactive Waste Canisters

	HLW Canisters[a] Existing[d] / Best Estimate	**Potential Range of HLW Canisters**
West Valley	275 / 275	not applicable[b]
Hanford	0 / 10,713	9,746–12,100
Idaho National Laboratory (Calcine)	0 / 3,328	1,190–11,200
Idaho National Laboratory (Electro-chemical processing)	0 / 102	82–135
Savannah River Site	3059 / 7,560	7,560–9,450
Total	3334 / 21,980	18,900–33,200[c]

NOTE: [a]Hanford canisters are 2 ft × 14.76 ft; all others 2 ft × 10 ft.
[b]All West Valley canisters currently exist.
[c]Rounded to nearest 100.
[d]Existing: current as of May 2011, except Savannah River Site current as of Dec 2010.

Source: after Carter et al. 2011b.

Projected Inventory of U.S. Department of Energy High-Level Radioactive Waste—The Savannah River Site currently has the only operating aqueous reprocessing facility in the U.S.: H Canyon. It is estimated that an additional 17,000 m^3 of liquid high-level radioactive waste may be generated with continued operations (until approximately 2019).

At Hanford and the Savannah River Site the liquid high-level radioactive waste currently in storage will be vitrified and placed in high-level radioactive waste canisters. Although the final waste form is known, the final number of high-level radioactive waste canisters cannot be estimated definitely because of potential changes in waste loading and processing.

At the Idaho National Laboratory several options were considered for ultimate disposal of the calcine. Alternatives included direct disposal, vitrification, or hot isostatic pressing to compress the calcine into a volume reduced monolithic waste form. A Record of Decision issued December 2009 stated that the U.S. Department of Energy will use the hot isostatic pressing technology to treat the calcine for final disposal in a geologic repository.

The Idaho National Laboratory is also pursuing an electro-chemical process for treating the 60 MTHM of sodium bonded used nuclear fuel from the Idaho National Laboratory, Hanford and Fermi I. The process has been demonstrated and used to treat about 4 MTHM of sodium bonded used nuclear fuel to date. The high-level radioactive waste generated from this process is converted into an iron based alloy and a glass bonded mineral.

Table 3-5 shows the projected number of high-level radioactive waste canisters that exist and that are estimated to be produced based on the proposed treatment technologies.

The vitrified high-level radioactive waste at the Savannah River Site is stored in below grade concrete vaults, called Glass Waste Storage Buildings, containing support frames for vertical storage of 2,262 high-level radioactive waste canisters. The Savannah River Site currently has two Glass Waste Storage Buildings constructed and a third planned. The high-level radioactive waste canisters at West Valley are currently stored in a shielded cell in the former reprocessing plant. Alternate interim storage options are being considered at West Valley to allow decommissioning of the reprocessing facility. Hanford has constructed the first canister storage facility and plans to construct more as needed when the Waste Treatment and Immobilization Plant becomes operational.

Characteristics of U.S. Department of Energy High-Level Radioactive Waste—The total high-level radioactive waste curie inventory for each of the generating sites, projected to 2017, is shown in Table 3-6. The baseline of 2017 is based on the date for waste acceptance used in U.S. Department of Energy (2008b), U.S. Department of Energy (2002b) and U.S. Department of Energy (2007). The total radionuclide inventory activity, by radionuclide, for each of the four sites has been tabulated in detail (Carter et al. 2011c, Appendix F). The Hanford site data in these tables do not include the Cs/Sr capsules pending a final disposal decision. This inventory was developed based on estimates provided by the sites. Although there may be some variation in the number of canisters produced by the sites that have not completed waste treatment, the total projected activity is as shown.

Table 3-6. High-Level Radioactive Waste: Total Curie (Ci) Inventory for Each of the Generating Sites, Projected to 2017

	Radioactivity (Ci)				
	Hanford	Savannah River	West Valley	Idaho	Total
Nuclide Total	1.3×10^8	9.54×10^8	1.46×10^7	2.58×10^7	1.13×10^9

The U.S. Department of Energy (2008b) used the “projected maximum” inventory on a per canister basis for the high-level radioactive waste curie content supplied by the Savannah River Site. The use of the “projected maximum” on a per canister basis resulted in a conservatively estimated total curie content for the Savannah River Site that is approximately twice the actual Savannah River Site tank farm inventory.

3.2.3 Waste from Reprocessing Commercial Light Water Reactor Used Nuclear Fuel

Reprocessing methods for existing and future commercial light water reactor used nuclear fuel vary in complexity and maturity of development of the technical aspects of implementation. In addition, the recovery of valuable material or the lessening of potential environmental impact of the waste disposition activities typically results in more complex processes. Several sources of uncertainties exist including (1) the expected timeline of at least twenty years for implementation, (2) technical advances during the implementation period, and (3) changing environmental regulations, which can place greater (or lesser) demands on the waste capture and treatment processes. Differing assumptions were used throughout the evaluations and cautious use of the results, especially comparisons, was recommended.

These results presented by Carter et al. (2011c) indicate that there may be limited benefit to reprocessing light water reactor fuel relative to associated volume reduction. However, the radionuclide content and decay heat production characteristics of the wastes differ significantly from those of bare fuel. These characteristics, together with the volume, have a significant role in management of these materials. Future research on advanced waste forms could lead to higher waste loading densities and a reduction in the volume of high-level radioactive wastes that would be generated from reprocessing.

The volumes of compacted hulls and hardware projected from the boiling water reactor fuels are higher than from pressurized water reactor fuels on a unit basis due to the lower ratio of fuel to structural materials. The quantity of the metal alloy wastes projected from aqueous reprocessing is about half the quantity projected from electro-chemical processing. This is due to inclusion of discarded electrode baskets and process crucibles in the metal waste stream. Projections of borosilicate glass quantity from the co-extraction process are limited by decay heat (14,000 watts per canister) when young used fuel is processed. Projections of borosilicate glass quantity from the co-extraction process when used fuel is older than 30 years are limited by molybdenum trioxide solubility. In these cases, the mass, volume, and containers per metric ton are a constant regardless of the age of the used nuclear fuel processed, although the decay heat continues to decline with fuel age.

Many of the fission product waste forms included in this study significantly exceed 1,500 watts per canister, which was a design limit used in U.S. Department of Energy (2008b). Additional decay storage time prior to disposal or a more dilute waste form may be required to meet this design limit, if applicable.

Projected Inventory and Characteristics of Commercial Used Light Water Reactor Fuel High-Level Radioactive Waste— Reprocessing technologies for used uranium oxide fuel may be broadly classified as aqueous and electro-chemical, neglecting the more limited dry technologies such as Volox (Strachan et al. 2009; Goff and Simpson 2009), which can be used as steps within the reprocessing methods discussed below. Aqueous processes generally involve dissolving used fuel in acid, and use various other steps (e.g., solvent extraction, precipitation, ion exchange) to separate fractions for waste and reuse as fuel. Intensive research and development has been conducted on reprocessing, especially the aqueous methods.

To support evaluations of the impact of reprocessing light water reactor used nuclear fuel, Carter et al. (2011c) examined three aqueous reprocessing methods and one electro-chemical reprocessing method:

1. **Co-Extraction—**The co-extraction method represents the simplest and most technically mature aqueous reprocessing method evaluated. The process considered is similar to the current generation of reprocessing technologies. It starts with dissolution of the uranium oxide fuel and solvent extraction of U and Pu. The U and Pu are recovered together; no pure Pu is separated. Afterwards, the principle fission product wastes including the minor actinides (Np, Am, Cm) are combined with the undissolved solids and recovered Tc into a single borosilicate glass waste form. (As proposed, this process is similar but not identical to the COEX™ process developed and used by the Areva company in France.)

2. **New Extraction—**New extraction is an advanced aqueous process that recovers all of the transuranic elements for re-use. After dissolution and solvent extraction of U with Pu, key features of the Transuranic Extraction (TRUEX) process (Schulz and Horwitz 1988) and the Trivalent Actinide Lanthanide Separation by Phosphorus-based Aqueous Komplexes (TALSPEAK) process (Weaver and Kappelmann 1964) are employed for complete transuranic element recovery and purification. The principle fission product wastes are combined with the undissolved solids and separated Tc into a single borosilicate glass waste form (Carter et al. 2011c). (This process has some similarity to, but is different from, the new extraction process proposed by the Energy Solutions company.)

3. **Uranium Extraction—**Uranium extraction (UREX) is an advanced aqueous process that also recovers and separates all of the transuranic elements including Pu and, in addition, separates the fission product waste components into three segments. The Fission Product Extraction (FPEX) process (Law et al. 2007) is added to separate the Cs, Sr, Ba, and Rb, which are converted to a solid ceramic waste form. The Tc and undissolved solids are combined with a portion of the zirconium-based cladding hulls and stainless steel hardware to form a metal alloy waste form, and the remaining fission products are converted to a borosilicate glass (Carter et al. 2011c). UREX is the most complex of the three aqueous processes evaluated.

4. **Electro-Chemical—**The electro-chemical process is a dry process using electro-chemical reactions in conductive molten salt baths to separate the transuranic elements from the other constituent metals and fission products in irradiated metal fuel from fast reactors. In this process the fission products are split between three waste streams. Elements that are more noble (as measured by electro-chemical potential) than U, such as used fuel cladding and noble metal fission products, remain as metals and are incorporated into a metal alloy waste form. Elements less noble than U are converted to chloride salts. The lanthanide elements (i.e., fission products) are selectively removed from the molten salt by electrolysis and converted to a lanthanide glass. Excess salt is removed from the bath by absorption in zeolite, which is bonded (i.e., fused) with a glass fraction to make the final waste form.

In each of the above processes, gaseous radionuclides including ^{129}I, ^{85}Kr, ^{14}C, and ^{3}H can be captured (e.g., using cold traps or chemical scrubbers) and converted to waste forms suitable for disposal, or released to the atmosphere if present in sufficiently small amounts, and if the health and safety of workers and the public can be demonstrated.

As an example of the types and characteristics of waste that could be generated with these reprocessing methods, the volumes of fission product wastes using the co-extraction and new extraction technologies are compared to the volumes from a sodium fast reactor (Carter et al. 2011c, Appendix D); an abstracted version is shown in Table 3-7 and Table 3-8 for pressurized water reactor and boiling water reactor fuels, respectively. The comparison looked at different burn-ups and ages of the used fuel. The volumes of fission product wastes are presented, together with the number of containers, and the decay heat of the material in the container. The container size was 0.61m in diameter by 4.6-m long.

Table 3-7. Comparison of the Volumes of Fission Product Wastes from Reprocessing Used Pressurized Water Reactor Fuel to Those from a Sodium Fast Reactor

Pressurized Water Reactor	**Burn-up**					
	20 GWd/MT		**40 GWd/MT**		**60 GWd/MT**	
	Age (yr)		**Age (yr)**		**Age (yr)**	
	5	**30**	**5**	**30**	**5**	**30**
Co-Extraction						
Volume (m^3/MT)	0.09	0.07	0.19	0.12	0.30	0.18
Containers per MT	0.07	0.05	0.14	0.09	0.23	0.13
Decay Heat (with container)	14,000	7,766	14,000	8,667	14,000	8,667
New Extraction						
Volume (m^3/MT)	0.06	0.06	0.12	0.12	0.17	0.17
Containers per MT	0.05	0.05	0.09	0.09	0.13	0.13
Decay Heat (with container)	12,124	2,911	13,111	3,157	13,851	3,233
Sodium Fast Reactor						
Volume (m^3/MT)	0.13	0.12	0.19	0.19	0.25	0.25
Containers per MT	0.09	0.09	0.14	0.14	0.19	0.19
Decay Heat (with container)	7,162	3,240	10,020	4,095	11,832	4,505

Table 3-8. Comparison of the Volumes of Fission Product Wastes from Reprocessing Used Boiling Water Reactor Fuel to Those from a Sodium Fast Reactor

Boiling Water Reactor	Burn-up					
	15 GWd/MT		30 GWd/MT		50 GWd/MT	
	Age (yr)		Age (yr)		Age (yr)	
	5	30	5	30	5	30
Co-Extraction						
Volume (m^3/MT)	0.07	0.04	0.13	0.09	0.25	0.14
Containers per MT	0.07	0.05	0.14	0.09	0.23	0.13
Decay Heat (with container)	14,000	9,457	14,000	8,999	14,000	9,499
New Extraction						
Volume (m^3/MT)	0.04	0.04	0.08	0.08	0.13	0.13
Containers per MT	0.03	0.03	0.06	0.06	0.1	0.1
Decay Heat (with container)	15,191	3,362	13,318	3,369	13,638	3,320
Sodium Fast Reactor						
Volume (m^3/MT)	0.11	0.11	0.16	0.16	0.23	0.21
Containers per MT	0.09	0.08	0.12	0.12	0.17	0.16
Decay Heat (with container)	5,717	2,594	8,621	3,673	10,458	4,220

3.3 Natural Barrier Characteristics

As seen in Figure 2-1, the natural barrier system consists of the host rock, within which the geologic disposal facility is developed, and the other geologic units surrounding it. The features of a natural barrier system are discussed in Section 2.2.2.1. The subsections below provide an overview of the processes that affect a generic natural barrier system, followed by a description of how the natural barriers of each of the four disposal options are impacted by different processes.

3.3.1 Summary of Natural Barrier System Processes

Processes affecting the natural barrier in a generic disposal system include flow, geochemical processes, and radionuclide transport. If radionuclides are released from the engineered barriers, radionuclides may be transported in the flowing groundwater either in solution (dissolved) or in suspension, bound to very small particles known as colloids. Advection, diffusion, dispersion, sorption, and colloid transport are all processes that can impact radionuclide transport.

Hydrologic Processes—Flow is the primary process for water entering the disposal system and either the primary or a significant contributor to the migration of radionuclides away from the disposal system. Above the disposal system, vertical flow in the bedrock beneath the surficial soil affects the rate of water movement below the soil–bedrock contact, especially in areas of thin soils. Horizontal flow is also a factor, especially in saturated transmissive units where horizontal flow rates are typically greater than vertical flow rates. While gravity is the primary driving force of flow in many potential repository settings, other possible driving forces include thermal processes, fluid density, overpressured conditions, and/or gas generation (e.g., from corrosion, radiolysis, or biodegradation). Below or adjacent to the disposal system, flow is an important mechanism contributing to the migration of radionuclides away from the disposal system. For those disposal system options that are dominated by advective processes, flow is the primary mode of transport. Diffusion is the other major mode of radionuclide migration.

Flow may occur through the rock matrix or through a fracture network in low permeability media. Porous-medium flow dominates in granular aquifers. When fractures are present they provide the dominant pathway for flow of water and transport of radionuclides. Radionuclides can diffuse out of the fractures and into the rock matrix, which contains a large reservoir of slower moving water, and which has a large surface area onto which radionuclide sorption can occur. In the absence of a fracture network, flow occurs only through the interconnected pore space in the rock. The location, magnitude, and direction of flow influence transport to the biosphere. The parameters that most affect flow in the natural system include the hydraulic gradient, hydraulic conductivity, and the rates of recharge and discharge. When fractures or faults are present in the host rock, they are often the dominant pathway for the flow of water entering the disposal system and the migration of radionuclides away from the disposal system. The presence of discrete fractures or faults controls the advective velocities and, therefore, the transport times to the accessible environment. The rate of water flow in fractures is influenced by such fracture properties as fracture frequency, fracture network interconnectedness, and fracture aperture. As the magnitudes of these properties increase, the effective hydraulic conductivity will also increase and result in an increase in flow.

Flow occurs at the surface and in the subsurface in a generic natural barrier system.

- Surface processes include infiltration, soil erosion and deposition, runoff, evaporation, and plant transpiration. By nature, they are highly site specific and are affected by seasonal and climate variations. Directly and indirectly, they impact the amount of water available to infiltrate into the formations beneath the surficial soils.

  Runoff can either decrease or increase the availability of water for infiltration depending on local conditions. Generally, steeper slopes have more runoff and less infiltration than more gentle slopes, but increased infiltration can occur, for example, under stream channels fed by runoff. The characteristics of the surficial soils and shallow bedrock also affect water retention and the time infiltrating water takes to pass below the root zone to become net infiltration (i.e., where it is not subject to further evapotranspiration processes). Evapotranspiration potentially removes a significant fraction of water from soil by evaporation and transpiration via plant root water uptake, and results in a reduction in the amount of water available to infiltrate into the subsurface rock beneath the surficial soils.

- Subsurface processes include those associated with flow in the rock matrix and/or in fractures and faults, which was discussed previously. The pattern of groundwater flow depends in part on whether one or more aquifer flow systems are present. In general, flow systems can be local, intermediate, or regional. Local flow systems are typically shallow with adjacent recharge and discharge areas; they are the most affected by seasonal variations in recharge. The system least affected by seasonal variations is the regional flow system, which extends much deeper and has discharge areas at the bottom of major drainage basins. Subsurface flow can also be affected by thermal and mechanical processes. These processes are less significant in the far field than in the near field because the primary zone of influence of these processes is the host rock near the disposal system and for a relatively short period of time compared to regulatory timeframes.

Geochemical Processes—Geochemical processes that occur in the natural system influence the barrier capability of the engineered system and influence the migration of radionuclides form the disposal system to the biosphere. Because many of these processes originate in the natural system, they are discussed in this section. Their impact on the engineered barrier system is discussed in Section 3.4.2. The physical and chemical environment that develops in the repository drifts is determined, in part, by the features and processes that comprise the portion of the natural system (geosphere) that lies above the repository.

In the natural system, local aqueous chemistry in the pore space or fractures of the host rock above the disposal system is controlled by the state of chemical equilibrium with respect to mineral phases, the effective surface areas of minerals in contact with water, the rates of mineral dissolution and precipitation, and the aqueous and gaseous fluxes and composition (Sandia National Laboratories 2007a). Liquid-gas phase exchange as percolating water moves toward the disposal system could have an effect on the composition of the water. Two potentially important processes to consider are evaporation (loss of water vapor) and CO_2 exchange. Mineral dissolution and precipitation lead to spatially heterogeneous changes in porosity of the rock. Porosity changes cause changes in hydrologic properties, which affect the occurrence and composition of the water entering the disposal system.

While the details of the geochemical interactions depend on site specific information, some general observations can be made. The chemistry of the water in the disposal system determines the corrosion mechanisms and rates of corrosion of engineered components. If the waste package has degraded to permit water to contact the waste form, this water can affect the chemical environment inside the waste package, influence the waste form degradation rate, and control the stability of radionuclide-bearing colloids and radionuclide solubility. The parameters that influence these processes include pH, ionic strength, composition of the pore water (e.g., carbonate levels), and oxygen partial pressure. The pH of water inside a breached waste package is controlled largely by mineral precipitation and dissolution reactions and ambient carbon dioxide levels. The chemistry of the water in the disposal system is determined from the chemistry of the water entering the disposal system as well as the processes and conditions (e.g. evaporation, condensation, relative humidity) that occur in the engineered barrier system and near field. The natural barrier system geochemistry also affects radionuclide dissolution, concentration, retardation, and colloid stability. These effects are discussed in the transport section, below.

Radionuclide Transport—Radionuclide transport is the mechanism that describes the migration of radionuclides from the disposal system through the engineered and natural barrier systems to the biosphere. Transport occurs in both unfractured rock and in the matrix and fracture network of fractured rock. When fractures are present in the natural system, they tend to dominate the transport of radionuclides for a number of reasons:

- High fracture permeability (compared to matrix permeability)
- Limited fracture pore volumes
- Limited fracture–matrix contact areas
- Short contact times between the radionuclide-carrying liquid phase and the rock matrix, where most of the radionuclide sorption occurs

Radionuclides may be transported as dissolved species in water or attached to suspended particle (colloids) in the water. Radionuclide transport in the natural system depends on the geologic material, hydrologic conditions, and geochemistry along the pathways of groundwater flow. There are a number of processes that contribute to the transport of radionuclides. These include advection, diffusion, dispersion, and sorption, and in the case of colloid transport the processes of filtration, straining, and stability also contribute.

The parameters associated with the processes that affect radionuclide transport include specific discharge, sorption (distribution coefficients), colloid retardation factors, dispersivity (longitudinal, horizontal transverse, and vertical transverse), radioactive decay constants, and fracture transport parameters (effective flow porosity flowing interval spacing and porosity, and effective matrix diffusion coefficient). The flowing interval porosity for fractured rock is defined as the volume of the pore space through which large amounts of groundwater flow occur relative to the total volume of the rock.

- **Advection**—Advection is a principal transport mechanism for both dissolved and colloidal radionuclides in the natural barrier system and thus has a direct impact on the barrier capability of the natural system. The advective flux depends on the hydrogeologic characteristics of the water-conducting features as well as the water flow rates through these features. The location, magnitude, and direction of flow influences transport to the accessible environment. Advection is characterized by the water velocity. Three parameters are typically combined to determine the velocity: hydraulic conductivity, hydraulic gradient, and effective porosity. In fractured rock the effective porosity is termed the flowing interval porosity. Effective porosity is that fraction of the porous medium through which water flow occurs. The advective velocity is typically determined as the specific discharge divided by effective porosity. Specific discharge is the volumetric groundwater flow rate per unit cross-sectional area. The presence of fractures or higher permeability strata or facies may significantly reduce the effective porosity relative to the total porosity of the medium.
- **Diffusion—**Diffusion, like advection and dispersion, is an important component of the natural system. Diffusion can occur in the matrix or in the fractures (if present). Diffusion is a process in which diffusing particles move in a manner similar to Brownian motion through both mobile and immobile fluids. Diffusion is a Fickian process by which species move from high- to low-concentration regions. The process of diffusion typically results in slower migration of radionuclides than with advection. The magnitude of diffusion depends on the free water molecular diffusion coefficient for individual constituents and the characteristics of the transport path through which the diffusing species passes. The diffusion coefficient of a species in the matrix is less than the free water diffusive coefficient because of the increased tortuosity of the matrix relative to free water. In a fractured system, diffusion of dissolved radionuclides into and out of matrix regions of slowly moving groundwater is an important process affecting radionuclide transport through the saturated zone. Dissolved radionuclides will diffuse from water flowing in the fractures into the rock matrix, as well as from flowing water in pores between rock grains in the matrix into pore spaces within the rock grains or between the rock grains (e.g., within clay lenses). The process of matrix diffusion is dependent on flowing interval spacing, sorption, matrix porosity, and effective diffusion coefficient for the radionuclide being analyzed. Because advective transport is significantly slower in the matrix than in the fractures, matrix diffusion can be a very efficient retarding mechanism, especially for moderately to strongly sorbing radionuclides, due to the increase in rock surface accessible to sorption. Radionuclides may also diffuse from the matrix back into the fractures. The direction and rate of diffusion depends on the relative concentration of the diffusing solute in the fractures and matrix.
- **Dispersion**—Dispersion is the mixing of a solute in flowing groundwater. Hydrodynamic dispersion combines mechanical dispersion, caused by localized velocity variations, with molecular diffusion, and is proportional to the concentration gradient. The net effect of these variations is to spread radionuclides over ever larger volumes as time and distances increase. Longitudinal dispersion causes some radionuclide mass to migrate either faster or slower than the average velocity along the groundwater flow trajectory. Transverse dispersion (both horizontal and vertical) causes solute plumes to widen perpendicular to the flow direction over time and distance. Longitudinal and transverse dispersions result in reduced concentrations within a solute plume relative to

concentrations that would be present in the absence of dispersion. The spreading and dilution of radionuclides that results from these heterogeneities is important to transport in the natural system. Dispersion tends to spread transient radionuclide pulses that may be released to the natural system. Dispersivity values, often separated into longitudinal dispersivity and transverse dispersivity, are needed to describe the dispersion process. Longitudinal and transverse dispersion occurs due to heterogeneity in permeability within the natural system. When transport is dominated by advection, dispersion becomes more important in spreading out radionuclide concentration.

- **Sorption—**Sorption removes a portion of the dissolved species from the mobile liquid phase and transfers it to the solid phase. The solid phase includes the immobile rock matrix, immobile colloids, and mobile colloids. Some radionuclides released from the repository are sorbed more readily than others. Several radionuclides (e.g., Sr, Cs, Pu, Ra, Am, Np, and U) that are important contributors to the total inventory can be retarded by sorption in the natural system. Some radionuclides are not sorbed at all (e.g., I, Cl, for some oxidation states, Tc). The sorption of these radionuclides in combination with radioactive decay prevents the movement or substantially reduces the rate of movement of these radionuclides from the repository to the accessible environment. Sorption reactions are chemical reactions that involve the attachment of dissolved chemical constituents onto solid surfaces. Sorption can be reversible or irreversible. Although these reactions can be complex, they typically are represented in transport calculations by a constant reversible sorption distribution coefficient, k_d. Sorption behavior is dependent on the characteristics of mineral surfaces in the rock units through which water flows in the saturated zone, the chemistry of the groundwater in the saturated zone, and the sorption characteristics and concentrations of each element. Variations in groundwater chemistry may influence complexation of radionuclides with other aqueous species, which can, in turn, impact the sorption of radionuclides onto the aquifer materials. Distribution coefficients tend to lump together multiple equilibrium and kinetic reactions and are specific to the conditions under which they were measured (e.g., pH, ionic strength, temperature, fluid-to-rock ratio, among others). Therefore, they provide only an approximate representation of the potential for contaminant retardation.

  The effects of sorption can be quantified in terms of a retardation factor, R_f (Freeze and Cherry (1979, Equation 9.14):

$$R_f = 1 + \frac{\rho_b k_d}{n} \qquad \text{Eq. 3-1}$$

  where:

  ρ_b = Bulk density of porous medium [M/L^3]

  k_d = Distribution coefficient [L^3/M]

  n = Porosity [L^3·L^3]

  The retardation factor provides an indication of the travel time of a sorbed radionuclide along a travel pathway relative to the travel time of a non-sorbing radionuclide (a non-sorbing radionuclide has k_d = 0 and R_f = 1). Radionuclides with k_d values of 10 mL/g or greater (which corresponds to $R_f \geq 100$ for typical rock densities and porosities) will move at less than 1% of the velocity of non-sorbing radionuclides.

- **Colloid Transport**—In general colloids and any associated radionuclides travel at a faster velocity than the average velocity of the water in which they are suspended. This is because they are restricted from experiencing the slow water velocities that exist near the rock walls along the flow path. Radionuclide-bearing colloids transported to the natural system may include (1) natural colloids, typically clay or silica; (2) waste-form colloids resulting from degradation of used nuclear fuel or high-level radioactive waste glass; (3) iron oxyhydroxide colloids resulting from degradation of the waste package, and (4) colloids from buffer materials in the engineered barrier system. These colloids are grouped into two types: those formed from hydrolysis of dissolved radionuclides (often called true colloids), and colloidal particles of other materials with attached radionuclides (called pseudocolloids). In some conditions, the generation and mobilization of colloids are important to radionuclide transport because they provide a transport mechanism to sorbing radionuclides that may otherwise not be present. Colloids provide sorbing radionuclides a mechanism of becoming partitioned into a mobile solid phase (the colloid) that moves with the water, as compared to sorbing onto an immobile solid phase.

Colloid transport in the natural system depends on the geologic material, hydrologic conditions, and geochemistry along the pathways of flow. Several processes influence the barrier capability of the natural system by affecting the rate of movement and concentration of radionuclide bearing colloids. These processes include colloid stability due to changing ionic strength, advection of colloids, diffusion of radioactive colloids into the rock matrix, sorption of radionuclides onto colloid surfaces, filtration of colloids by the rock, and straining of colloids.

- *Stability*—In order for radionuclide-bearing colloids to affect transport in the saturated zone, the concentration of colloids in suspension must be stable over the timeframe of transport and must carry significant amounts of radionuclides. The stability of colloids is dependent on ionic strength and pH, and is controlled by electrostatic and chemical processes at the colloid-solution interface that determine the balance between attractive and repulsive forces between adjacent colloids. Higher ionic strengths weaken repulsive forces between colloids, causing colloidal suspensions to become unstable and to agglomerate. Agglomeration is favored, when the pH is near the pH of zero point of charge of a particular colloid. Under unsaturated conditions, high temperatures and low humidity favor high-ionic strength solutions that would destabilize colloids causing them to settle out of suspension.

- *Sorption*—Sorption describes a combination of physical and chemical interactions between dissolved radionuclides and the solid phases. Sorption removes a portion of the dissolved species from the mobile liquid phase and transfers it to the solid phase. The solid phase includes immobile colloids and mobile colloids. Sorption onto immobile colloids results in retardation of radionuclide transport in the natural system. However, sorption onto mobile colloids can enhance radionuclide transport. Most sorbing radionuclides sorb onto colloids reversibly (i.e., they have measurable desorption rates and can be entirely desorbed from colloids). However, Pu and Am can sorb either reversibly or irreversibly onto colloids. Irreversibly sorbed Pu and Am are either embedded within waste form colloids (e.g., smectite colloids formed by degradation of high-level waste glass), or are so strongly sorbed onto colloids formed in the waste package environment (e.g., iron oxyhydroxide colloids formed by corrosion of waste packages) that there is no possibility of desorption over typical transport timescales through the natural system. The implication of this is that once attached to colloids these radionuclides will migrate through the natural system faster than their dissolved counterparts.

  Radionuclides that are reversibly sorbed onto colloids will transport both on colloids and in the aqueous phase, and their transport characteristics are a combination of the transport characteristics of both the solute and colloids. Colloid-facilitated transport of reversibly sorbed radionuclides depends on colloid transport parameters, mobile colloid mass concentrations, and radionuclide sorption onto both colloids and the immobile rock matrix. High colloid concentrations and large radionuclide distribution coefficients onto colloids (relative to the rock matrix) favor colloid-facilitated radionuclide transport. Aqueous chemical conditions play an important role in the process of radionuclide sorption onto colloids. The behavior of the mineral surface is primarily controlled by pH and ionic strength. The behavior of the sorbate is primarily controlled by its oxidation state, pH, and the partial pressure of carbon dioxide (pCO_2) and to a lesser extent on ionic strength.

- *Filtration*—Filtration is the physical attachment or detachment of colloids onto an immobile rock surface. It may be reversible or irreversible. Filtration of colloids occurs on both fractures and matrix but is very important in reducing the migration of colloids when flow is primarily through a fracture network. Filtration of colloids results in retardation of the movement of radionuclides associated with colloids. Transport through the rock matrix is relatively slow and sorption or filtration is much more likely to occur because of greater contact areas and longer contact times between water and rock.
- *Straining*—Colloid straining can also affect the distribution and transport of colloids. Straining mechanisms can be classified according to the relative size of the colloid as conventional straining, where the colloid is larger than the pore throat, and film straining, where the colloid is larger than the thickness of the adsorbed water film coating the grains of the rock. Colloid removal by straining is strongly dependent on colloid size and dependent on water flow rate if unsaturated conditions prevail. When the water saturation of the pore space is lower than a critical saturation value, colloids can only move in the thin film of water that lines the grain boundaries.
- *Diffusion*—In general, matrix diffusion is less significant for colloids than for solutes because the large size of the colloids reduces the colloid diffusion coefficient, reduces the number of colloids entering the matrix because of straining or pore size exclusion.

3.3.2 Natural Barrier System Processes for the Four Disposal Options

3.3.2.1 Generic Salt Natural Barrier

Much of the information relating to generic disposal in salt in this section comes from Hansen and Leigh (2011) and the *Compliance Certification Application for the Waste Isolation Pilot Plant* (U.S. Department of Energy 1996a). Additional details can be found in these two documents.

Salt formations may be bedded or domal. Bedded salt is composed of halite (NaCl) with significant amounts of anhydrite ($CaSO_4$), polyhalite ($K_2MgCa_2(SO_4)_4{\bullet}2H_2O$), and clay minerals as minor phases. Brine exists in bedded salt in three forms: fluid inclusions, hydrous minerals, and grain boundary water. Owing to the characteristics and environments of the brine in salt, its migration occurs via three primary mechanisms: motion of the brine inclusions in a temperature gradient, vapor-phase transport along connected porosity, and liquid transport driven by the stress gradient. Thick (10s to 100s of meters) beds of relatively pure halite are often interspersed with thin (meters), nearly horizontal anhydrite interbeds. These interbeds are more permeable than halite by several orders of magnitude and represent potential pathways for migrating radionuclides through the natural system. This potential can be mitigated by locating a repository some vertical distance from these interbeds. Anhydrite is more brittle than halite and may develop horizontal fractures that can enhance its horizontal permeability when exposed to fluid pore pressures near lithostatic levels. Such pressure could conceivably come from gas generation in the repository from corrosion or radiolysis of water present.

Disposal of used nuclear fuel and high-level radioactive waste in salt is attractive because the natural barrier capability of salt formations is very robust. Salt, particularly in the intact halite portions of the formation, is characterized by extremely low permeability (10^{-22} to 10^{-24} m^2), very low porosity (0.1% to 1%), and reducing chemical conditions. Additionally the force exerted by the overburden results in creep consolidation of the surrounding salt encapsulating the waste in the disposal rooms and restoring the DRZ to near intact conditions. Other positive attributes for salt disposal include:

- Salt is easily mined
- Salt has high thermal conductivity
- Salt formations can be stable, with many having existed underground for millions of years
- Salt deposits can be found in geologically stable regions

Hydrologic Processes and Radionuclide Transport—The low permeability that characterizes a salt natural system, even considering the presence of anhydrite, is primarily responsible for its strong barrier functions of containment and limited release. Advective transport pathways may exist depending on the stratigraphy, presence, and location of anhydrite interbeds relative to the repository. However, even if contaminated brine from a salt repository finds an advective path (e.g., through an anhydrite interbed), its migration to the biosphere will be extremely slow. In addition, some radionuclide sorption will occur in these interbeds, although not as much as expected in clay or shale. Typically, the efficacy of a salt repository does not rely on radionuclide sorption as a significant mitigation for release. Distribution coefficients for radioelements of interest in a dilute brine, consistent with conditions in an anhydrite interbed, are presented in Table 3-9. The low porosity greatly limits the volume of brine that is potentially available, and the low permeability of salt and anhydrite greatly limits its movement.

Table 3-9. Salt k_d Values for a Dilute Brine at 25°C

Element	Distribution Type	k_d (mL/g)	Source
U	Uniform	0.2 (min); 1 (max)	Lappin et al. (1989); McKinley and Scholtis (1992); Muller et al. (1981); Tien et al. (1983)
Pu	Uniform	70 (min); 100 (max)	
Np	Uniform	1 (min); 10 (max)	
Am	Uniform	25 (min); 100 (max)	
Th	Uniform	100 (min); 1000 (max)	
Tc	Uniform	0 (min); 2 (max)	
Cs	Uniform	1 (min); 20 (max)	
Ra, Sr	Uniform	1 (min); 80 (max)	
Ac, Cm	Log-uniform	5 (min); 500 (max)	McKinley and Scholtis (1992) (k_d values reduced by a factor of 10 to account for the high salinity of brine.)
C	Uniform	0 (min); 0.6 (max)	
Nb, Pd	Constant	0.1	
Pa	Log-uniform	1 (min); 500 (max)	
Sb	Constant	10	
Se	Uniform	0.2 (min); 0.5 (max)	
Sn	Uniform	2 (min); 10 (max)	
Zr	Log-uniform	3 (min); 500 (max)	
Cl, I, Pb	Constant	0 (no sorption)	

Source: Clayton et al. 2011, Table 3.1-7.

The stability barrier function of salt is demonstrated by the presence of thick salt beds over geologic timescales. These beds remain unaltered by external natural events, e.g. seismic or igneous activity. The depths of the deposited salt preclude them from being disturbed by glaciation or other natural surface processes.

The WIPP provides a reasonable analogue for a generic repository in bedded salt. In the absence of external events, safety assessment calculations conducted for the successful certification (U.S. Department of Energy 1996a) and recertification (U.S. Department of Energy 2004a) of the WIPP demonstrated that there is no pathway for movement of radionuclides out of the isothermal salt repository except diffusion, which is extremely limited. The only potential releases are from human intrusion scenarios where flooding of the repository and/or pressure increases lead to direct releases to the surface through boreholes. Because the salt surrounding the repository is nearly impermeable, pressures in the repository can exceed hydrostatic pressure and approach lithostatic pressures by the processes of gas generation from corrosion of waste packages and radiolysis and from consolidation of the salt under lithostatic loading of the overburden. The gas-generating processes however, require the presence of brine. Consequently, the volume of brine and its ability to flow contribute to the barrier capability of salt by limiting brine availability and subsequent gas generation and pressure rise. The regulations governing the WIPP (40 CFR Part 191) prescribe including these surface releases during human intrusion in determining cumulative release. In contrast, other assessments may not require consideration of surface releases during or after human intrusion.

As understood from the compliance basis for the WIPP, for an undisturbed repository at low temperatures, brine comes in contact with the waste primarily by flowing through or from the DRZ. A very small amount can also flow through the seal system. Most of the brine that enters the repository originates in the DRZ. A small portion migrates along a thin anhydrite interbed located within a few meters of the repository. The permeability of undisturbed halite is too low to permit any significant migration of brine. A thermally driven repository could respond differently, but in either setting, there is a strong possibility for complete healing of the damage zone because of the mechanical response of salt to stress differences.

Despite the large initial hydraulic gradient between the far field and the excavated repository, the low permeability of the natural system restricts the flow of brine toward the repository. Over time this pressure difference will be reduced in an effort to re-establish an equilibrium condition. The reduction in the gradient will not come as a result of brine flow, but rather from an increase in repository pressure from gas generation and consolidation. Beauheim and Roberts (2002) performed an evaluation of Salado Formation hydrology and hydraulic properties. They conclude that:

> *On the time scale of the operational period of WIPP (decades), the far field lacks the capacity to fill all of the newly created porosity in and around the repository, much less pressurize it to near lithostatic pressure. After WIPP is closed, far-field flow toward the repository will continue, but the overall "healing" of the formation around the repository (closure) and compaction of the crushed-salt backfill will act to reduce both the hydraulic gradient and the porosity present near the waste. Thus, the amount of brine that ever comes into contact with waste will be controlled by the relative rates at which brine flow and repository closure occur.*

Thus, after repository closure as salt creep closes a disposal room, the stress gradient decreases, any pre-existing fractures or those generated by the excavation or gas pressurization heal, and crushed salt backfill (if present) reconsolidates. As a result, conditions in a repository would evolve to significantly limit brine flow to the waste disposal areas. The impact of elevated temperatures from high-activity waste on the processes occurring in a salt repository are not expected to negatively impact the capability of the salt natural system and may result in improved capabilities associated with the engineered barrier system (e.g. more rapid consolidation and healing of the DRZ and creation of a dry-out zone in the near field) (Section 4.2.3.2.1).

In regard to WIPP, the National Academy of Sciences (1996) concluded that:

> *Provided it is sealed effectively and remains undisturbed by human activity, the committee finds that the WIPP repository has the ability to isolate TRU waste for more than 10,000 years. The geologic stability and isolation capability of the Salado Formation, which consists of bedded salt, are the primary factors leading to this finding.*
>
> *The only known possibilities of serious release of radionuclides appear to be from poor seals or some form of future human activity that results in intrusion into the repository. The committee anticipates that the consequences of such human intrusion can be reduced based on available engineering design options and on improved understanding to be obtained from ongoing scientific studies.*

Geochemical Processes—Geochemistry in a repository located in salt is controlled in part by the interactions between the salt formation brine and the waste, packaging, and emplacement materials. The in-situ pH of brines is slightly acidic (about 6.0 to 6.5). Mineral components of the salt formation buffer the pH to their in-situ values. Salt formation brines tend to have high concentrations of sodium, calcium, and chloride. Lesser amounts of sulfate and carbonate are likely to be present. Some brines also have high magnesium concentrations. These interactions determine the pH, oxidation/reduction conditions, gas fugacities, and the dominant aqueous species present. Because of the extremely low permeability little or no communication exists between the natural system brines and the engineered barrier system. As a result the geochemical conditions in the disposal regions are primarily determined by the near field and within the repository rather than the natural system. The geochemistry is also influenced by temperature but the thermal period is short lived and also restricted to the engineered barrier system (Section 4.2.3.2.1).

The key factors that establish the concentrations of dissolved radionuclides in addition to the amount of brine are:

- Redox chemistry
- Complexation
- Intrinsic and pseudo colloid formation and stability

If radionuclides were to be released into the natural system, the geochemical environment would be dominated by reducing conditions. Such an environment limits corrosion of the waste canisters, contributing to their long-term ability to isolate waste although no long-term performance of the engineered barrier system is required as the salt natural system is sufficiently robust to provide the necessary barrier functions. A reducing environment also limits solubility and/or increases the sorption of most radionuclides, further delaying the migration of these radionuclides to the biosphere. For a given oxidation state, complexing agents present define the solubility of a radionuclide. The solubility of radionuclides is influenced by the geochemistry as migration through the natural system occurs. Solubility helps determine radionuclide concentrations that influence transport of dissolved or colloidal radionuclides. The oxidation state-specific solubilities of key radioelements in brines have been established for isothermal repositories like WIPP and Asse (Hansen and Leigh 2011). Solubilities for radioelements of interest in a concentrated brine, consistent with chemically-reducing conditions, are presented in Table 3-10. Complexants are either in the pre-emplacement environment or exist in the waste form. In addition, some of the key radioelements in their expected oxidation states tend to form colloids (intrinsic colloids) or strongly associate with colloids potentially present because of clay seams that intersect the salt formation (pseudocolloids). The high ionic strength of the brines would be expected to limit the stability and transport of these colloids.

Table 3-10. Elemental Solubilities of Radionuclides in a Concentrated Brine at 25°C

Element	Distribution Type	Dissolved Concentration (mol/L)	Source
U	Triangular	4.89E-08 (min); 1.12E-07 (mode); 2.57E-07 (max)	Wang and Lee (2010)
Pu	Triangular	1.40E-06 (min); 4.62E-06 (mode); 1.53E-05 (max)	
Am	Triangular	1.85E-07 (min); 5.85E-07 (mode); 1.85E-06 (max)	
Np	Triangular	4.79E-10 (min); 1.51E-09 (mode); 4.79E-09 (max)	
Th	Triangular	2.00E-03 (min); 4.00E-03 (mode); 7.97E-03 (max)	
Tc	Log-triangular	4.56E-10 (min); 1.33E-08 (mode); 3.91E-07 (max)	
Sn	Triangular	9.87E-09 (min); 2.66E-08 (mode); 7.15E-08 (max)	
Ac, Cm	Constant	5.85E-07	Clayton et al. (2011, Appendix C)
Cl	Constant	4.20	
Nb	Constant	1.60E-05	
Pa	Constant	1.51E-09	
Pd	Constant	4.00E-04	
Sb	Constant	6.30E-05	
Se	Constant	2.00E-05	
Zr	Constant	1.00E-10	
C, Cs, I, Pb, Ra, Sr	N/A	Unlimited solubility	

Source: Clayton et al. 2011, Table 3.1-4.

3.3.2.2 Generic Clay Natural Barrier

Much of the information on generic clay disposal comes from Hansen et al. (2010). Additional details can be found in that reference. The natural system in clay disposal exhibits strong positive attributes for permanent waste isolation including uniformly low hydraulic conductivity, low diffusion coefficients, good retention capacity for radionuclides (e.g., Blümling et al. 2007), and conditions that are chemically reducing. Other positive attributes include plasticity, fracture sealing or healing, but these contribute more to the engineered barrier system (Section 3.4). Comparison studies of programs in various countries (U.S. Nuclear Waste Technical Review Board 2009) have concluded that the long-term performance of an engineered barrier system is unimportant to the safety case for disposal in clay media. Thus, the waste package, though integral to the disposal concept, can readily be engineered to meet design or manufacturing objectives.

Clays, as used in this report, represent sediments in the spectrum from poorly unconsolidated clay to lightly indurated argillaceous media, with approximately 50% clay content and low permeability. They have small particle size and substantial sorption capability for holding water or ions (Bates and Jackson 1980). Clay mineral particles have a large ratio of surface area to volume and have high ion-exchange capacity and low permeability. The most common clay minerals are kaolinite, montmorillonite, illite and chlorite, each of which is actually a group of similar minerals. Clay media may contain significant fractions of water-soluble salts, calcite, chemically reducing minerals such as pyrite, and organic material.

More specifically, the features of a repository in clay relevant to the natural system include:

- **Geologic Stability**—Contributes to the barrier function of stability. Clay/shale media have persisted tens of millions to hundreds of millions of years in almost all geologic provinces in the U.S., and have mostly remained in the same state that was acquired soon after deposition (Gonzales and Johnson 1984).
- **Mechanical Behavior**—Contributes to the barrier function of limited release. High clay content is needed to ensure low permeability and plasticity. The more clay-rich, plastic, less indurated, and less

fissile clays are preferred for repository purposes. An important characteristic of clay formations is the anisotropy, due to their sedimentary origin. It is visible in many parameters, such as principal stresses, elasticity modulus, or hydraulic conductivity. Clay formations and deep clays in particular all exhibit creep, which depends strongly on the presence of water. This phenomenon is therefore most important in plastic clays. The ductility of clay results in the self-sealing of fractures around openings and is a primary favorable feature of clay media. Fractures formed by excavation and heating will close and seal during repository operation and during the first few hundred years after closure, limiting fluid movement.

- **Hydrologic Properties** (e.g., low permeability and porosity)—Contributes to the barrier functions of containment and limited release. The intrinsic permeability of clay is typically on the order of 10^{-19} m^2 or less corresponding to a hydraulic conductivity of 10^{-12} m/s or less. The presence of overpressured fluid in the Opalinus clay (estimated to have a head as much as 100 m greater than a water column to the ground surface (Nagra 2002) gives strong evidence of low permeability. For this excess head to persist over geologic time signifies that advective transport is not significant in the repository timeframe. Such low permeability results from the impermeable nature of clay in general, and also from a lack of permeable fractures open to flow.

  Depending on site specifics there is potential for faults, fracture zones, or other structural features to exist in some clays. No evidence of natural fractures has been found in the potential siting region in Boom Clay; however, fractures have been observed in quarry or pit excavations of some clay formations being investigated in Europe (Arnould 2006), where fracture spacings from a few millimeters to one meter are observed in surface exposures (where stress release and weathering have occurred)., These fractures are closed by confining pressure at depth and are not generally observed. Such fractures have apparently never before been open, and have never conducted significant fluid flow, because they are not locally altered by exposure to water and have no filling mineral deposits.

- **Favorable Chemical Environment**—Contributes to the barrier functions of containment and limited release. A repository excavated in a clay formation would have a reducing environment that would limit corrosion of the waste canisters, contributing to their long-term ability to isolate waste. These reducing conditions also enhance radionuclide sorption and limit radionuclide solubility for many radionuclides of importance. The large specific surface area associated with clay further enhances the capacity of clay to absorb radionuclides, limiting their transport.

  The pH of the waters in clay systems would tend to be buffered by reactions with carbonate phases, to slightly alkaline values, and to a lesser extent by surface protonation/deprotonation reactions of clay minerals. These characteristics, with a high capacity for buffering cation and anion content, make bentonite backfill a favorable material for use in borehole and shaft seals to maintain a constrained water composition (Bradbury and Baeyens 2003).

Hydrologic Processes—Hydrology impacts the barrier functions of containment and limited release. Disposal of used nuclear fuel and high-level radioactive waste in a clay formation is expected to provide effective long-term (> 1 million years) isolation of radionuclides from the biosphere. In part this is due to slow fluid movement. Fluid movement in clay or shale is slow because of very low hydraulic conductivity. Water infiltrates the natural clay barrier at a very low rate owing to the very low permeability of the material. This infiltration is initially considered to obey the principles of unsaturated flow, but as the clays swell in the presence of water, the flow is further impeded and the mechanism of flow changes (Delage et al. 2010). An exhaustive study by the “Clay Club” (Organisation for Economic Co-operation and Development 1996) describes the basic physical and chemical processes that combine to control the flow of water, gas, and solute through clay media. Much is known about these processes, but without a specific site in mind this study selects from a range of properties encountered in clay repository studies reported internationally.

Radionuclide Transport—The primary mode of transport through the clay natural system is expected to be diffusive over 1,000,000 years or longer. Radionuclide transport in the far-field host rock is limited by low permeability on the order of 10^{-19} m^2 or less (Hansen et al. 2010). Application of advective-diffusive transport modeling to coupled hydrogeochemical transport (Hansen et al. 2010) indicates maximum extents of radionuclide transport on the order of tens to hundreds of meters, or less, in a million years. Under the conditions modeled, a clay repository could achieve total containment, with no releases to the environment in undisturbed scenarios.

Diffusion—Diffusion dominates the transport of dissolved radionuclides in a disposal system in clay. Representative measured values of D_a in bentonite (representing clay media and backfill/buffer or sealing materials) are:

- Non-sorbing, uncharged species (tritium): 10^{-9} m^2/s (Bradbury and Baeyens 2003)
- Anions such as iodide: 3 to 7 × 10^{-11} m^2/s (Lee et al. 1994)
- Moderately sorbing cations such as neptunyl and Cs^+: 10^{-12} to 6 × 10^{-11} m^2/s, depending on the dry density of the bentonite (e.g., Bradbury and Baeyens 2003)
- More strongly sorbing cations: 10^{-12} m^2/s

Advection—Advective groundwater flow through clay host rock is negligible. Hansen et al. (2010) investigated the potential for advective transport in clay/shale. Properties typical of clay/shale were used: e.g., permeability of 10^{-19} m^2 (corresponding to hydraulic conductivity of approximately 10^{-12} m/s), effective porosity of 10% of total porosity, and an upward hydraulic gradient is assumed of 0.001 (pore velocity of 3.15×10^{-7} m/yr). The corresponding advective travel time of an unretarded dissolved radionuclide through a 150-m-thick clay unit is greater than 100 million years, which is consistent with the long-term stability of natural hydrogeochemical conditions in clay formations. Sensitivity analysis shows that the predicted dose at 1,000,000 years is relatively insensitive to the hydraulic gradient, as long as the advective pore velocity is less than approximately 10^{-4} m/yr.

Solubility—Table 3-11 identifies the likely solubility-limiting phases and provides estimates of in-situ dissolved concentrations for some of the key radioelements for the environment associated with a disposal system in clay/shale.

The relatively low solubility of UO_2 (uraninite) under reducing conditions would favor stabilization of used fuel rods. Dissolution would occur at conditions close to equilibrium with uraninite. Because uraninite is the stable uranium phase, negligible oxidative degradation of the waste form would be expected. For example, a natural analogue study of uranium fixation in a Tertiary argillite found that uraninite was the principal secondary uranium phase formed (Havlova et al. 2006). When contacted by water, fuel rods would have diminished thermodynamic drive to dissolve, thus slowing the release of actinides and fission products from the fuel matrix. Yet even if fuel rods were to instantly degrade to the thermodynamically stable actinide oxides, each of these actinide phases has low solubility that would limit contributions to the source term from the isotopes of Am, Ac, Cm, Np, Pa, Pu, Tc, and Th.

It is less clear whether I, Ra, and Sr would form solubility-limiting solids. If clay fluids contained appreciable sulfate, $SrSO_4$ and $RaSO_4$ might form to limit dissolved Sr and Ra levels. These phases would be more likely to form in the more sulfate-rich solutions found in some bentonite backfill materials (Bradbury and Baeyens 2003). Dissolved carbonate might also lead to the formation of $SrCO_3$. Low values for dissolved Ra concentration (2×10^{-11} mol/L) were based on a solid solution model for Ra within barium sulfate (Schwyn and Wersin 2004) considering solutions likely to form in bentonite backfill, for the Swiss repository concept in the Opalinus clay. In that study dissolved Sr was limited to 2×10^{-5} mol/L. There are possible solid solution phases that could incorporate Sr, similar to Ra, such as calcite and barite.

Kinetically limited reduction of Se by FeS_2 or possibly organic matter is proposed as a solubility limiting mechanism (Maes et al. 2004). Radioiodine should be reduced to highly soluble iodide given sufficient electron donors from steel waste containers and organics in the clay. No limiting concentrations are set for I, Cs, and Sr in Table 3-11. These radioelements would be controlled by their inventories and the slow dissolution rates of the waste forms, for example the UO_2 matrix grains of the used fuel.

Table 3-11. Elemental Solubilities of Radionuclides in Clay Formations

Element	Dissolved Concentration[a] (mol/L)	Dissolved Concentration[b] (mol/L)	Solubility-Limiting Phase[b]	Notes[b]
Am, Ac, Cm	4.0×10^{-7}	1×10^{-6}	$AmOH(CO_3)$	Am solubility is used as proxy for chemically similar Ac and Cm.
C	2.3×10^{-3}	None	None	Likely limited by calcite growth
Cl	None	No value reported		
Cs	None	None	None	No solubility limiting phase
I	None	None	None	No solubility limiting phase
Nb	2.0×10^{-7}	No value reported		
Np	4.0×10^{-9}	6×10^{-12}	NpO_2	
Pa	1.0×10^{-6}	6×10^{-12}	PaO_2	Np solubility is used as proxy for chemically similar Pa.
Pb	4.0×10^{-6}	No value reported		
Pd	4.0×10^{-7}	No value reported		
Pu	2.0×10^{-7}	5×10^{-6}	$Pu(OH)_4$	
Ra	1.0×10^{-7}	None	$RaSO_4$	Possible solid solution with $BaSO_4$
Se	5.0×10^{-10}	No value reported		
Sb	None	No value reported		
Sn	1.0×10^{-8}	No value reported		
Sr	None	None	Possibly $SrCO_3$, $SrSO_4$	Possible solid solution
Tc	4.0×10^{-9}	3×10^{-31}	TcO_2	
Th	6.0×10^{-7}	6.0×10^{-8}	$Th(OH)_4$	
U	7.0×10^{-7}	1.0×10^{-9}	UO_2	
Zr	2.0×10^{-8}	No value reported		

NOTE: [a] Representative of the Callovo-Oxfordian formation. (Source: Clayton et al. 2011, Table 3.3-23, based on Andra 2005d)

[b] Calculated for T=100°C and pH=7.0 using the PHREEQC code version 2.12.03 and the thermo.com.V8.R6.230 database from Lawrence Livermore National Laboratory (except for the T=25°C TcO_2 solubility product and enthalpy, which came from a separate thermodynamic database (SNL 2008c, Section 6.3.7.5)). The solution assumed 50 mmol S, 50 mmol bicarbonate, and calcite saturation. Values assume chemically reducing conditions. (Source: Hansen et al. 2010, Table 2.5-1)

Sorption—Clays have a large capacity to retard the migration of radionuclides through sorption and ion exchange. Distribution coefficients for radioelements of interest in the chemical environments associated with clay formations are presented in Table 3-12, based on data provided for the Callovo-Oxfordian clay in France (Andra 2005d). As noted in Section 3.3.1, distribution coefficients tend to lump together multiple equilibrium and kinetic reactions and are specific to the physical and chemical conditions under which they were measured. Therefore, they provide only an approximate representation of the potential for contaminant retardation. Nevertheless, k_d values are useful in examining controls on radionuclide transport.

Elements with k_d values of 0 (e.g., Cl, I, Se) do not sorb and will therefore move at the velocity of the fluids that carry them. Elements with k_d values of 10 or greater will move at less than 1% of the velocity of the non-sorbing radionuclides. Table 3-12 reinforces that sorption will sharply limit the transport of most radionuclides from clays and shale.

Table 3-12. Clay k_d Values

Element	k_d (mL/g)
Am, Ac, Cm	50,000
C	0.4
Cl, I, Sb, Se, Sr	0
Cs	388
Nb	4,810
Np, Pu	900
Pa, Ra	1,000
Pb	160
Pd	805
Sn	16,100
Tc, Zr	1,150
Th, U	8,000

Source: Clayton et al. 2011, Table 3.3-23, based on Andra 2005d.

Colloid Transport—Although clays and shale produce colloids, they are strongly filtered as they transport through the natural system.

Example—Hansen et al. (2010) investigated the transport of radionuclides from a disposal system in clay/shale. Their results indicate concentrations of ^{129}I, ^{238}U, ^{236}U, ^{79}Se, ^{234}U, ^{233}U, and ^{135}Cs above 10^{-10} mg/L at the clay/aquifer interface assumed to be 150 m distant from the repository. The peak concentrations of ^{129}I and ^{238}U are 2×10^{-4} mg/L at about 100,000 years and 4×10^{-6} mg/L at 1,000,000 years respectively. Hansen et al. (2010) also investigated release to the biosphere using a stylized groundwater pumping scenario. They reported a total dose of 0.01 mrem/yr over 1,000,000 years. This is believed to be a bounding value because of the conservative nature of the calculations: all waste is assumed to instantly degrade and dissolve inside the waste packages; all waste is assumed to be pressurized water reactor assemblies; unlimited availability of moisture for waste form degradation and transport is assumed; no sorption on degraded waste package materials is allowed.

3.3.2.3 Generic Granite Natural Barrier

Much of the information on generic granite disposal comes from Mariner et al. (2011). Additional details can be found in that reference. Unlike generic disposal in salt, clay, or deep boreholes, the total system barrier capability for a mined generic repository in fractured granite depends largely on waste package preservation. Whereas the natural setting may attenuate sorptive or redox-sensitive waste species that are released from the repository, the more mobile species (e.g., ^{129}I) may be transported rapidly by fracture flow. However, disposal of used nuclear fuel and high-level radioactive waste in a repository deep in a granite formation is expected to provide effective long-term ($>10^6$ years) isolation of radionuclides from the biosphere because of mechanical, hydrologic, and chemical conditions favorable to waste form containment. In crystalline rock, waste packages are preserved by the high mechanical stability of the excavations, the diffusive barrier of the buffer, and favorable chemical conditions. The buffer is preserved by low groundwater fluxes, favorable chemical conditions, backfill, and the rigid confines of the host rock. The barrier aspects of the engineered system are discussed in Section 3.4.

More specifically, the features of a granite repository relevant to the natural system include:

- **Geologic Stability**—Contributes to the barrier function of stability. Large quantities of homogeneous granites are found in regions of low seismic activity and are known to have been stable for millions of years.
- **Mechanical Properties**—Contribute to the barrier function of stability and containment. Granite exhibits great strength, mechanical stability, and rock homogeneity, all of which enhance excavation stability. Many mined tunnels/caverns in granitic rocks have remained intact for centuries without additional support. An excavation in suitable granite formations would be stable and long-lasting (Office of Crystalline Repository Development 1983). Mechanical stability would contribute to worker safety during the preclosure and operational periods. Waste packages and other engineered barriers (buffers and backfill) would be protected from shear stresses and rockfall during both the operational and postclosure periods.
- **Hydrologic Properties** (e.g., low permeability and porosity)—Contribute to barrier functions of containment and limited release. Intact granite is known for its low permeability; meaningful groundwater flow occurs only through fractures. A repository excavated in a suitable granite formation would have low rock permeability, which limits groundwater flow. Low interconnected porosity, in combination with the already low porosity of granite, limits groundwater flow in granite. Advective flow of groundwater can lead to the enhanced migration of radionuclides away from the repository and towards the biosphere. In the granite natural system, lighter freshwater is found near the surface and heavier saline water is found at depth, which is a stable arrangement that tends to reduce vertical groundwater flow. Chemical age dating indicates that groundwater at the depths of granite repository sites is very old, which supports the hypothesis that saline water at depth does not mix readily with surface waters. Hence, radionuclide migration from the repository to the biosphere would be slowed.
- **Favorable Chemical Environment**—Contributes to the barrier functions of containment and limited release. A repository excavated in a suitable granite formation would have a reducing environment that would limit corrosion of the waste canisters, contributing to their long-term ability to isolate waste. These reducing conditions also increase radionuclide sorption and limit radionuclide solubility for many radionuclides of importance. The high salinity of groundwater typically observed in deep granitic rocks inhibits colloidal transport of radionuclides and could present some resistance to upward flow due to its likely higher density than overlying waters.
- **Hydrology**—Impacts the barrier functions of containment and limited release. Granite formations often contain fractures. Fractures, when present and if the fracture network is interconnected to the biosphere, are the dominant pathway for the flow of water to the repository and the migration of radionuclides to the biosphere. In a suitable granite formation, fractures should be small and sparse in the vicinity of the repository, and the bulk hydraulic conductivity of the granite should be low (approximately 10^{-10} m/s or lower). Rock fractures are known to dissipate in number with depth and may be insignificant at repository depths.

Hydrologic Processes—For granite at depths of at least 400 m, the hydraulic conductivity is commonly in the range of 10^{-10} to 10^{-13} m/s (Mariner et al. 2011). Sparsely fractured granite at this depth can have low flow (kinematic) porosities on the order of 10^{-4} and diffusion porosities on the order of 10^{-3} (Posiva 2008; SKB 2006a). The flow porosity is the volume of rock through which water flows relative to the bulk volume of rock. For granite, essentially all flow occurs through interconnected fractures, and the flow porosity is often referred to as the interconnected fracture porosity. The much larger diffusion porosity in granite consists of both matrix and dead-end fracture porosity and is connected to the flow porosity. The diffusion porosity is often referred to as the matrix porosity. The extremely low fracture porosity of deep granite causes the velocity of water in the interconnected fractures to be orders of magnitude higher than the specific flux. Consequently, advection may dominate solute transport in the interconnected fractures while diffusion dominates transport in the matrix. Based on the low hydraulic conductivity of granite host rock and the likely low hydraulic gradient in the vicinity of a well-sited repository, groundwater flow at repository depth would be quite limited. Transmissive fractures at this depth may have an average spacing of more than 100 m as found at Forsmark (SKB 2011). As a result, complete resaturation of the backfill and buffer after repository closure may not occur for hundreds or thousands of years (SKB 2011).

Geochemical Processes—The chemical environment within a granite natural barrier system contributes to the barrier functions of containment and limited release. Some of the processes contributing to this function include the effects of reducing conditions on canister corrosion rates, radionuclide sorption, and radionuclide solubility. In addition; the high salinity of groundwater typically observed in deep granitic rocks would inhibit radio-colloid transport and could present some resistance to upward flow due to its likely higher density than overlying waters.

At the depth of a potential repository in granite (approximately 500 m), brackish groundwater typically saturates the fractures and interconnected pores. Na-Ca-Cl solutions predominate at this depth with total dissolved solids in the range of 1 to 10 g/L or higher (Mariner et al. 2011). Granitic waters at these depths are typically not modern. Isotope data at Forsmark, for example, indicate that water of meteoric origin does not occur below about 200 m (SKB 2006a). At Olkiluoto, mixing models indicate that water at a depth of 500 m is approximately 50% formation water, 10% melt water from the Weichselian glaciation (the most recent glacial period from approximately 110,000 to 10,000 years ago), and 40% water from pre-Weichselian Quaternary glacial cycles (Posiva 2010b). The maximum fraction of Weichselian melt water at Olkiluoto (~50%) is observed at a depth of approximately 150 m. These data indicate that water at repository depths in granite is not well connected to the biosphere.

Reducing conditions are important to repository performance because they can prevent rapid oxidation of waste canisters, decrease aqueous solubilities of many redox sensitive radionuclides, increase sorption capability of many radionuclides, and can limit waste form degradation rates. The redox conditions at the depth of a potential repository in granite are reducing. There are two primary reasons for this. First, there is little mixing of infiltrating waters to depths of 400 to 500 m. Second, there is an abundance of oxygen-consuming reactants below the surface. At shallow depths, oxygen is typically consumed by microbial degradation of organic carbon. At Olkiluoto, iron oxyhydroxides are observed in fractures only in the top few meters (Posiva 2010b). Below this depth, pyrite and other mineral sulfides are present and react with oxygen, producing sulfate. At approximately 300 m at Olkiluoto a spike in the hydrogen sulfide concentrations indicates reducing conditions are strong enough to reduce sulfate to sulfide. Below 300 m, aqueous concentrations of both sulfate and sulfide drop and the concentration of methane rises. Methane is another strong buffer against the downward transport of oxygen because methanotrophs will use available oxygen to oxidize methane.

Studies in Sweden and Canada also indicate reducing conditions in granite formations. All groundwater samples from Äspö and Stripa were shown to contain dissolved Fe(II) despite prolonged periods of oxygen inflow into the tunnels (SKB 2006a). Microbial respiration was shown at Äspö to consume infiltrating dissolved oxygen in the first 70 m of a major fracture zone even after construction of a tunnel through the fracture zone at that depth caused a 20-fold reduction in the mean residence time (Banwart et al. 1999). In Canada, all Eh measurements from groundwater samples from four research areas in the Canadian Shield indicate redox potentials at or below the Fe(II)/Fe(III) boundary (Gascoyne et al. 1987).

Radionuclide Transport—Transport in a fractured media such as granite can occur through the fractures or the matrix.

<u>Advection and Diffusion</u>—Transport through interconnected fractures is generally dominated by advection while transport into the connected porosity of the matrix is dominated by diffusion. In fractured granite, the matrix porosity acts to retard solutes relative to solute velocities within the fracture porosity. This diffusion-related retardation effect applies to the transport of both sorbing and non-sorbing solutes.

Thus, for fractured granite with a fracture porosity of 10^{-4} and a matrix porosity of 10^{-3} solutes would spend an average of ten times as much time within the stagnant domain of the matrix than within the fractures (assuming the solutes are non-sorbing or equally retarded by sorption within each domain).

Sorption—Radionuclide sorption in granite occurs on fracture walls, fracture minerals, and in the granite matrix. Table 3-13 lists distribution coefficients adopted by the Finnish repository program for modeling sorption in the far-field granite at Olkiluoto (Posiva 2010b). These values are based on laboratory measurements and consider the site-specific rock types, fracture minerals, and estimated dilute/brackish groundwater composition in the granite natural system. Table 3-13 also lists distribution coefficients for sorption in far-field granite reported by the Swedish repository program (Carbol and Engkvist 1997; SKB 2010).

Table 3-13. Granite k_d Values

Element	k_d (mL/g) [b]	k_d (mL/g) [c]	k_d (mL/g) [d]
Am, Ac[a], Cm[a]	40	0.6 – 383	3,000
C	0	0	1
Cl, I	0	0	0
Cs	50	0.03 – 3.5	50
Nb	20	1.1 – 353	1,000
Np, Th	200	2.8 – 984	5,000
Pa	50	6.8 – 518	1,000
Pb	Not reported	2.1 - 310	0
Pd	1	1.2 – 2,210	10
Pu	500	0.6 – 383	5,000
Ra	200	0.04 – 1.5	20
Sb	Not reported	Not reported	0
Se	0.5	0.003 – 3.5	1
Sn	1	45 – 558	1
Sr	5	0.00004 – 0.3	0.2
Tc	50	2.8 – 984	1,000
U	100	2.8 – 984	5,000
Zr	200	4.5 – 102	1,000

NOTE: [a] k_d values for Ac and Cm are set equal to those of chemically similar Am.
[b] Representative of dilute/brackish groundwater in far-field granite at Olkiluoto, Finland (Source: Posiva 2010b, as reported in Mariner et al. 2011, Table 2-3).
[c] Representative of granite at Forsmark, Sweden (Source: SKB 2010, Table 6-89).
[d] Representative of far-field granite in Sweden (Source: Carbol and Engkvist 1997).

Solubility—Radionuclide solubility is an important process contributing the ability of radionuclides to be transported because for many radionuclides their dissolved concentrations are limited. Mariner et al. (2011) reported on the solubility limits associated with radionuclides in waters consistent with waters common to generic granite natural system (Table 3-14). C, Cl, Cs, I, Sr, and Pb do not have limiting concentrations. The concentrations of these radioelements are expected to be controlled by their inventories, instant release fractions, and/or the slow dissolution rates of the waste forms. Actual solubilities of these elements are moderate to high in this chemical environment.

Colloid Transport—With respect to colloid transport in granite the high salinity of the groundwater is expected to significantly limit colloid stability in the granite host rock (Sandia National Laboratories 2007b). Depending on the depth of the repository and other site specific conditions, the effects of glaciation could result in changes to the chemistry from dilute groundwater. The potential for colloidal transport would need to be considered in a site-specific safety assessment.

Table 3-14. Elemental Solubilities of Radionuclides Solubilities in Granite Formations

Element	Solubility-Limiting Phase	Dissolved Concentration[a] (mol/L)	Notes
Am, Ac, Cm	$Am(OH)_3$	6×10^{-6}	Ac, Cm assumed analogous to Am
Np, Pa	$Np(OH)_4$	1×10^{-9}	Pa assumed analogous to Np
Nb	$Nb(OH)_5$	4×10^{-5}	Posiva (2010b, Table 1-9)
Pd	$Pd(OH)_2$	3×10^{-6}	Posiva (2010b, Table 1-9)
Pu	$Pu(OH)_4$	2×10^{-7}	
Ra	$RaSO_4$	1×10^{-6}	(SO_4^{2-}) fixed at 10^{-3} mol/L
Sb	$Sb(OH)_3$	1×10^{-7}	
Se	$FeSe_2$	4×10^{-8}	
Sn	SnO_2	3×10^{-8}	
Tc	TcO2:2H2O(am)	3×10^{-8}	
Th	$Th(OH)_4$	4×10^{-7}	
U	UO_2	4×10^{-10}	
Zr	$Zr(OH)_4$	2×10^{-8}	Posiva (2010b, Table 1-9)
C,Cl,Cs,I,Sr,Pb	None	None	

NOTE: [a]Calculated for T=25°C and pH=7.5 using the PHREEQC code version 2.14.2 and the thermo.com.V8.R6.230 database from Lawrence Livermore National Laboratory, except where noted. The solution assumed 0.3 M NaCl, 0.05 M $CaCl_2$, 10^{-3} m Na_2SO_4, and 10^{-7} atm H_2 (g).

Source: Mariner et al. 2011, Table 2-5.

3.3.2.4 Generic Deep Borehole Natural Barrier

Much of the information on generic granite disposal comes from Brady et al. (2009) and Arnold et al. (2011). Additional details can be found in those references. A proposed design of the deep borehole concept for disposal of used nuclear fuel and high-level radioactive waste can be found in Arnold et al. (2011). The approach involves drilling a borehole (or array of boreholes) into crystalline basement rock to a depth of about 5,000 m, emplacing waste canisters containing used nuclear fuel and high-level radioactive waste in the lower 2,000 m of the borehole, and sealing the upper 3,000 m of the borehole. Sealing of the upper part of the borehole would be done with a series of compacted bentonite seals, cement plugs, cement seals, cement plus crushed rock backfill, and bridge plugs.

Numerous factors indicate that deep borehole disposal of high-activity waste is inherently safe. Waste in the deep borehole is several times deeper than for typical mined repositories, resulting in greater natural isolation from the surface and near-surface environment. Several lines of evidence indicate that groundwater at depths of several kilometers in continental crystalline basement rocks has long residence times and low velocity. High salinity fluids have limited potential for vertical flow because of density stratification, which also prevents colloidal transport of radionuclides. Geochemically reducing conditions in the deep subsurface limit the solubility and enhance the retardation of key radionuclides.

More specifically, the features of deep borehole disposal relevant to the natural system include:

- **Geologic Stability**—Contributes to the barrier function of stability. Large quantities of homogeneous crystalline rocks of suitable thickness and depth are found throughout the U.S. in regions of low seismic activity and are known to have been stable for millions of years
- **Mechanical Behavior**—Contributes to barrier functions of stability and containment. Crystalline rocks such as granites are particularly attractive for borehole emplacement. They exhibit great strength, mechanical stability, and rock homogeneity, all of which enhance drilling stability, waste emplacement, and containment. Waste packages and other engineered barriers (buffers and backfill) would be protected from shear stresses and rockfall during both the operational and postclosure periods. In addition, high overburden pressures contribute to sealing of some of the fractures that provide transport pathways.

- **Hydrologic Properties** (e.g., low permeability and porosity)—Contribute to barrier functions of containment and limited release. Crystalline rocks at depths in excess of 3 km are known for their low permeabilities, poorly connected transport pathways, and lack of open fractures at the depths involved. Fluid movement is inhibited by low porosities (< 1%), very low permeabilities (10^{-16} to 10^{-20} m^2), and the presence of convectively-stable, high ionic strength brines (≥ 150 g/L). The permeabilities of deep crystalline rock are roughly 10 orders of magnitude less than those of gravel aquifers. The porosities of deep crystalline rock are 10 to 40 times less. Deep crystalline rocks typically have low water content.

- **Favorable Chemical Environment**—Contributes to the barrier functions of containment and limited release. The geochemical behavior (solubility, sorption, colloidal behavior, etc.) of the projected waste inventory in the deep borehole environment sets limits on the stability of the used UO_2 fuel matrix and on radionuclide transport to the biosphere. The chemical environment of the natural system surrounding the waste disposal section of a deep borehole is reducing. Geochemically reducing conditions in the deep subsurface limit the solubility and enhances sorption of many radionuclides in the waste, leading to limited mobility in groundwater.

  Fluids recovered from deep boreholes tend to be rich in sodium, calcium, and chloride. Lesser amounts of sulfate and carbonate are likely to be present. For the purposes of estimating radionuclide solubilities, a reasonable salinity is ~ 2-3 mol/L, pHs are 8-9 and the system Eh is ~ −300 mV (Anderson 2004). Oxygen tends to be scavenged, and the low redox state anchored, by the presence of reduced Fe and Mn in the basement rocks, resulting in lower solubility limits. Additionally, high ionic strength brines will limit the formation and movement of radionuclide-bearing colloids.

  Additionally, geothermal gradients are such that the temperatures at the bottom of the deep boreholes are expected to be above 100°C. Geochemically appealing features of deep boreholes are that the elevated temperatures of deep boreholes should stabilize the less soluble crystalline forms of radioelement oxide minerals, while high temperatures and high salinities will both favor the less soluble anhydrous forms of the oxide phases.

Hydrologic Processes—Hydrology impacts the barrier functions of containment and limited release. Disposal of used nuclear fuel and high-level radioactive waste in a deep borehole is expected to provide effective long-term (> 1 million years) isolation of radionuclides from the biosphere. Advective flow of groundwater can lead to the enhanced migration of radionuclides away from the repository and toward the biosphere. The physical transport of radionuclides away from the waste canisters at multi-kilometer depths would be limited by the following: low water content, low porosity and low permeability of crystalline basement rock, high overburden pressures that contribute to the sealing of transport pathways; and the presence of convectively stable saline fluids. Fluid flow is thought to occur primarily through discontinuous fractures. Basement rocks do not typically contain pressurized aquifers or other flow features that would produce significant upward flow gradients under ambient conditions (Brady et al. 2009). Low permeability and high salinity in the deep continental crystalline basement at many locations suggest extremely limited interaction with shallow fresh groundwater resources (Park et al. 2009), which is the most likely pathway for human exposure.

Groundwater in deep (>3,000 m) crystalline basement rocks of stable continental regions typically has chemical and isotopic characteristics that indicate it is very old. Chemical age dating indicates that groundwater at the depths of deep borehole disposal sites is very old, which suggests that saline water at depth does not mix readily with surface waters (Möller et al. 1997). Hence, radionuclide migration from the repository to the biosphere would be slow. Therefore, the most significant driving force for fluid flow and radionuclide migration away from a deep borehole is likely to be minor thermal-hydrologic effects from decay heat. Deep fluids also resist vertical movement because they are density stratified. The density stratification of groundwater would also oppose thermally induced groundwater convection from the waste to the shallow subsurface. In the natural system, lighter freshwater is found near the surface and heavier saline water is found at depth, which is a stable arrangement that tends to reduce vertical groundwater flow.

The heat generated from the waste emplaced in the borehole will cause fluid temperatures and pressures to rise in the vicinity of the waste. Thermal hydrologic calculations estimate the thermal pulse from emplaced waste to be small (less than 20°C at 10 m from the borehole, for less than a few hundred years), and to result in maximum total vertical fluid movement of ~100 m. (Brady et al. 2009). The elevated pressure will drive fluid away from the heated zone. The path of least resistance will be up the sealed borehole and adjacent disturbed host rock, where permeabilities are likely to be higher than that of the undisturbed bedrock. Model results from Brady et al. (2009, Figure 8) indicate that upward fluid flow in the heated borehole only persists for a relatively short period of time (from approximately 34 to 600 years in this example) after emplacement. The maximum pore velocity at the top of the waste zone was 0.662 m/yr, but upward flow in this area only occurs for the first approximately 180 years. Maximum pore velocity at the top of the basement domain peaks at 0.103 m/yr at about 150 years. Fluid movement is primarily caused by the local elevated pressures due to thermal expansion of the pore water. As the heat generation decreases, the temperature of the waste decreases and the fluid begins to contract, lowering pressure.

Permeability of the host rock near the borehole potentially could be enhanced by hydrofracturing resulting from the thermal expansion of fluid. This might increase the permeability in the host rock around the sealed borehole and provide a pathway for upward vertical flow and radionuclide migration toward the surface. This potential process was evaluated in Brady et al. (2009) and excluded. The hydrothermal modeling results suggest that comparable fluid pressures would not be achieved and that no hydrofracturing would occur by this process with maximum fracturing pressures of 40MPa compared to in-situ horizontal stress of 96MPa.

Radionuclide Transport—Potential transport pathways to the biosphere are long and would therefore involve extensive radioactive decay, dilution, formation of radionuclide-bearing phases, and retardation, given the impediments to vertical migration of radionuclides from several kilometers depth. Transport of radionuclides can be advective and diffusive. The primary mode of transport through the natural system surrounding a deep borehole disposal facility is expected to be due to the advection of dissolved radionuclides.

<u>Diffusion</u>—Brady et al. (2009) examined the potential for diffusive transport and given the depth of deep borehole disposal system concluded diffusion could be eliminated as a viable mode of radionuclide transport to the biosphere. Their bounding analysis showed that over 1,000,000 years diffusion migration distances would be limited to about 200 m.

<u>Advection</u>—While transport by advection is the dominant mode for radionuclide migration, it is still very slow. Thermal-hydrologic calculations (Brady et al. 2009) indicate that, except for an early window extending from the time of emplacement to ~150 years post-emplacement (in the borehole), and ~600 years (to the top of the basement), there will be limited vertical fluid flow to transport radionuclides towards the surface. Vertical transport velocities in the early flow window will be between 0.1 (basement) and 0.7 (borehole) m/yr. This means that total vertical fluid movement in, and adjacent to, deep borehole disposal zones should not exceed roughly 100 m.

Brady et al. (2009) also examined the hypothetical release to the biosphere via groundwater pumping though a nearby well intersecting an aquifer. The peak dose over 1,000,000 years occurs at 8,200 years. The total dose is negligibly small, 1.4×10^{-10} mrem/yr (~10 orders of magnitude below current criteria), and the only contributor to the dose is ^{129}I. This result is based on several bounding and conservative assumptions, such as (1) all waste is assumed to instantly degrade and dissolve inside the waste canisters, (2) all waste is assumed to be pressurized water reactor assemblies, and (3) no credit is taken for sorption or decay along the saturated zone transport pathway from the sealed borehole to the withdrawal well. More refined safety assessments may indicate lower doses, or later peak doses, or both.

Solubility—Table 3-15 provides estimates of dissolved radioelement concentrations at the depths of an environment characteristic of deep borehole disposal. The relatively low solubility of UO_2 under deep borehole conditions will favor stabilization of used fuel rods. When contacted by water, fuel rods will have diminished thermodynamic drive to dissolve, thus slowing the release of actinides and fission products from the fuel matrix. Yet even if fuel rods were to instantly dissolve to the thermodynamically stable actinide oxides, the solubilities of isotopes of Am, Ac, Cm, Np, Pa, Pu, Tc, Th, and U are very low, suggesting that aqueous releases of these radionuclides will be small.

Table 3-15. Elemental Solubilities of Radionuclides in Deep Boreholes

Element	Dissolved Concentration[a] (mol/L)	Dissolved Concentration[c] (mol/L)	Solubility-Limiting Phase[c]	Notes[c]
Am	6.5×10^{-9} (mode)[b]	1×10^{-9}	Am_2O_3	$AmOH(CO_3)$ would control Am solubilities if carbonate present.
Ac	6.5×10^{-9}	1×10^{-9}	Ac_2O_3	Am solubility is used as proxy for chemically similar Ac.
C	None	None	None	No solubility limiting phase
Cm	6.5×10^{-9}	1×10^{-9}	Cm_2O_3	Am solubility is used as proxy for chemically similar Cm.
Cl	4.2	No value reported		
Cs	None	None	None	No solubility limiting phase
I	None	None	Metal iodides?	See discussion
Nb	1.6×10^{-5}	No value reported		
Np	1.9×10^{-6} (mode)[b]	1.1×10^{-18}	NpO_2	Np solubility is used as proxy for chemically similar Pa.
Pa	1.9×10^{-6}	1.1×10^{-18}	PaO_2	
Pb	None	No value reported		
Pd	4.0×10^{-4}	No value reported		
Pu	3.56×10^{-14} (mode)[b]	9.1×10^{-12}	PuO_2	
Ra	None	None	$RaSO_4$	See discussion
Se	2.0×10^{-5}	No value reported		
Sb	6.3×10^{-5}	No value reported		
Sn	2.66×10^{-8} (mode)[b]	No value reported		
Sr	None	None	$SrCO_3$, $SrSO_4$	See discussion
Tc	1.33×10^{-8} (mode)[b]	4.3×10^{-38}	TcO_2	
Th	3.37×10^{-8} (mode)[b]	6.0×10^{-15}	ThO_2	
U	9.46×10^{-13} (mode)[b]	1.0×10^{-8}	UO_2	
Zr	1.0×10^{-10}	No value reported		

NOTE: [a] Representative of a chemically reducing brine at T=200°C. (Source: Clayton et al. 2011, Table 3.4-4)

[b] mode of a triangular distribution from Wang and Lee 2010

[c] Calculated for T=200°C and pH=8.5 using the PHREEQC code version 2.12.03 and the thermo.com.V8.R6.230 database from Lawrence Livermore National Laboratory (except for the 25°C TcO_2 solubility product and enthalpy, which came from a separate thermodynamic database (SNL 2008c, Section 6.3.7.5)). The solution assumed 2 M NaCl and Eh= -300 mV. (Source: Brady et al. 2009, Table 4)

It is less clear whether I, Ra, and Sr will form solubility-limiting solids. If deep fluids contain appreciable sulfate, $SrSO_4$ and $RaSO_4$ might form to limit dissolved Sr and Ra levels. Dissolved carbonate might also lead to the formation of $SrCO_3$. Radioiodine is a fission product that should become reduced to iodide given sufficient electron donors in the borehole domain. Unless iodide forms insoluble metal iodides, radioiodide levels in solution adjacent to the fuel will probably be set by the available inventory. Pending closer examination of sulfate, carbonate, and heavy metal contents of borehole fluids, no limiting concentrations are set for I, Sr, and Ra. Isotopes of C, Cs, and Pb also have no limiting concentrations, and Cl has a very large limiting concentration.

Sorption—Radionuclide sorption has rarely been measured at temperatures much greater than 25°C. Nevertheless, there is sufficient experimental data to suggest that most radionuclides released from the bottoms of deep boreholes will sorb to basement rocks, to overlying sediments, and to the bentonite used to seal the borehole. Table 3-16 provides a compilation of representative distribution coefficients, k_d's.

Table 3-16. Deep Borehole k_d Values

Element	Disposal Zone[a]	Seal Zone[b]	Upper Zone[c]
	k_d (mL/g)	k_d (mL/g)	k_d (mL/g)
Am, Ac[d], Cm[d]	5–500	300–29,400	100–100,000
C	0–0.6	5	0–2,000
Cl, Pb	0	0	0
Cs	5–40	120–1,000	10–10,000
I	0–1	0–13	0–100
Nb	1	10	10
Np, Pa[d]	1–500	30–1,000	10–1,000
Pd	1	5–12	4–100
Pu	1–500	150–16,800	300–100,000
Sr, Ra[d]	0.4–3	50–3,000	5–3,000
Sb	10	100	100
Se	0.2–0.5	4–20	1–8
Sn	2–10	17–50	50–700
Tc[e]	0–25	0–250	0–1,000
Th	3–500	63–23,500	800–60,000
U	0.4–500	90–1,000	20–1,700
Zr	3–500	100–5,000	100–8,300

NOTE: [a] k_d values for deep basement granite at T=100°C under chemically reducing conditions, reduced by a factor of 10 to account for sorption in a highly saline disposal zone. (Source: Clayton et al. 2011, Table 3.4-3, based on Brady et al. 2009, Table 5 and McKinley and Scholtis 1993)

[b] k_d values for bentonite seals at T=100°C under chemically reducing conditions. (Source: Clayton et al. 2011, Table 3.4-5, based on Brady et al. 2009, Table 5 and McKinley and Scholtis 1993)

[c] k_d values for sediments at T=25°C under less chemically reducing conditions that the seal and disposal zones. (Source: Clayton et al. 2011, Table 3.4-6, based on Brady et al. 2009, Table 5 and McKinley and Scholtis 1993)

[d] k_d values for Ac and Cm are set equal to those of chemically similar Am. k_d's for Pa are set equal to those of chemically similar Np. k_d values for Ra were set equal to those of somewhat chemically similar Sr.

[e] k_d values for Tc under reducing borehole conditions will likely be much greater than the zero values listed here which were measured under more oxidizing conditions.

As noted in Section 3.3.1, distribution coefficients tend to lump together multiple equilibrium and kinetic reactions, are specific to the conditions under which they were measured (e.g., pH, ionic strength, temperature, fluid-to-rock ratio, among others) and, therefore provide only a rough predictor of the potential for contaminant retardation (McKinley and Scholtis 1993; Bethke and Brady 2000). Nevertheless, k_d's are useful in examining controls over radionuclide transport. Elements with k_d's of 0 (for example, I, Cl, and Pb) won't sorb and will therefore move at the velocity of the fluids that carry them. Elements with k_d values of 10 or more will move at less than 1% of the velocity of non-sorbing radionuclides. Table 3-16 emphasizes that sorption will sharply limit the transport of most radionuclides from deep boreholes.

Colloid Transport—The transport of radionuclides in the form of colloids would be limited because the high ionic strength of the high salinity pore fluids at depth would destabilize the colloids causing them to leave suspension and settle out in the formation.

3.4 Engineered Barrier Characteristics

As seen in Figure 2-1, the engineered barrier system contains the waste form and waste package as well as the engineered features of the geologic disposal facility, which typically include buffer material, backfill, excavation liner, and/or seals. A general description of the engineered barrier system features and their functions is given in Section 2.2.3.1. Additional information regarding the potential waste types and waste packaging is provided in Section 3.4.1. Section 3.4.2 summarizes various processes relevant to the functioning of the engineered barrier system.

3.4.1 Potential Waste Forms and Waste Packages

Several types of waste require deep geologic isolation for permanent disposal: used nuclear fuel, high-level radioactive waste from reprocessing, and perhaps greater-than-Class-C waste from various sources. These types of waste and some finer distinctions are discussed in the following subsections.

3.4.1.1 Used Nuclear Fuel

Used uranium oxide fuel is a polycrystalline ceramic material, stable to high temperatures, and with the potential for slow degradation behavior in the disposal environment. Fuel assemblies in commercial light-water reactors are categorized by physical configuration into 22 classes: 16 for pressurized water reactors and 6 for boiling water reactors. Assemblies are of similar size within each class, and overall there are 134 individual fuel assembly types (Carter and Luptak 2010). The overall inventory of fuel in reactors and used fuel in storage is tracked by industry groups and the U.S. Department of Energy. Approximately 65,000 MTHM of used fuel were projected to have been discharged from reactors by 2010 (Table 3-1). The fission energy yield from the fuel (called "burn-up") determines the composition and heat output of used fuel. Whereas the average burn-up for fuel discharges projected through 2010 ranged from 33.3 GWd/MT (for boiling water reactors) to 39.6 GWd/MT (for pressurized water reactors), the burn-up for fuel presently being discharged is generally greater than 45 GWd/MT (Section 3.2.1). This means more energy is being produced for the same mass of used fuel, but it also means the used fuel comprising the waste form is hotter and contains more fission products and other radionuclides besides the initial uranium.

Cladding protects the fuel from degradation in the reactor, and the fuel is likely to be received for disposal at the repository with cladding intact. Cladding can protect the fuel from degradation in the repository also, especially in the event of a waste package breach during the period of elevated temperature, i.e., during the first few hundreds to thousands of years after emplacement. Cladding from commercial light-water reactors is generally made from Zircaloy (Type 1L is typical) a zirconium alloy that is chemically stable and resistant to corrosion. It can be damaged by internal pressurization of the fuel rods by fission-product gases; such damage is controlled by limiting the fuel temperature. Various measures have been developed for handling, storage, and packaging of the fuel to ensure that cladding does not get hot enough for such damage to occur (BSC 2005a; U.S. Department of Energy 2008b, Section 1.2). Cladding performance may be incorporated into performance assessments to realistically represent its contribution to waste isolation.

For the long waste isolation performance timescales, used uranium oxide fuel degrades, particularly if exposed to oxidizing water and aqueous or gaseous CO_2. The used fuel is exposed to an operating temperature of approximately 1700°C in the reactor, during which it undergoes physical changes due to heating, radiation damage, and the buildup of fission products (Bruno and Cera 2006). The used fuel is cracked, with lighter elements (consisting of fission products) concentrated in voids and the outer margins of the UO_2 matrix. A labile fraction of these fission products is released over time as the cladding slowly degrades. Many of the fission products have shorter half-lives (e.g., on the order of 10,000 years or less) and decay in the waste package before they are released. Release of other radionuclides occurs as the UO_2 matrix slowly degrades in the presence of water, accelerated by the activity of CO_2 or carbonate, and elevated temperature.

Used Fuel Dry and Wet Storage—Most used reactor fuel in the U.S. is presently stored in pools at the reactor sites. These pools were not designed with capacity to accommodate the used nuclear fuel that would be discharged over the life of nuclear power plants, especially since their licenses to operate have been extended, so they are filling to capacity with used fuel.

Large, off-site fuel pools are being used for used fuel management internationally. For example, the Central Interim Storage Facility for Spent Nuclear Fuel (CLAB) in Sweden is physically separated from any reactor, and will hold the spent fuel from Sweden’s entire nuclear energy industry for up to 100 years or until the fuel is cool enough and the packaging and repository facilities have been developed and licensed (SKB 2006b, Section 9.2). The commercial facility at Morris, Illinois is currently the only licensed storage facility in operation in the U.S. that is not located at a reactor site (Carter and Luptak 2010), and it uses “wet” pool storage.

As fuel pools in the U.S. are essentially full, dry storage systems are being used to off-load used fuel, and more than 1,100 dry storage canisters have been deployed at commercial plants (Carter and Luptak 2010). By 2020 the industry projects that more than 30,000 MTHM of used fuel will be in dry storage, in approximately 2,200 casks, distributed among commercial reactor locations in a number of different states (Carter and Luptak 2010). Dry storage systems demonstrate safe and economical fuel handling, canisterization (for those storage systems using canisters), and transport. All of these operations would be used in an integrated disposal system in the U.S., which would likely include centralized, large-scale dry storage either in storage casks or vaults, because much of the fuel at reactors is already loaded in dry storage canisters.

Containers for Storage and Transport—Used fuel at the reactor fuel pools is loaded into storage canisters, storage casks, or transport casks. All of these containers are loaded with fuel assemblies underwater, within the fuel pool. Storage and transport casks are loaded with bare fuel assemblies and moved respectively to surface dry storage. Canisters are distinguished from casks, and are lighter, relatively thin-walled stainless steel vessels with minimal radiation shielding. Canisters are loaded with used fuel, dried, sealed, and then immediately received into a transfer cask for movement to local, stationary storage casks or vaults. Canisters can be transported to other fuel management facilities (e.g., independent spent fuel storage facilities that may be located at other sites) without re-opening or re-packaging. These fuel handling and packaging operations have been demonstrated to be safe to workers and the public (Garrick 2003).

Canistered fuel in dry storage casks can be safely transported as indicated by the existence of licensed transportation casks or designs, and the safety record of used fuel shipments in the U.S. Used fuel canisters may also be suitable for disposal (in addition to storage and transport). The concept of a multi-purpose canister licensed for storage, transport, and disposal has been well studied, and was considered for the U.S. program in the early 1990s (U.S. Department of Energy 1994). More recently, a multi-purpose transportation-aging-disposal canister was proposed for handling most of the used fuel from the reactors to a repository (U.S. Department of Energy 2008b, Section 1.2.1).

Disposal Packaging for Used Fuel—Fuel transported to the repository in canisters is likely to be placed into disposal casks (“overpacks”) with characteristics that will depend on the requirements of the disposal system. Repository concepts with host media such as salt, clay, or the crystalline basement which have very low permeability, may have simpler requirements on disposal casks as indicated by preliminary performance assessments (Clayton et al. 2011; Hardin et al. 2011a). Multi-purpose canisters (licensed for disposal as well as storage and transportation) could be emplaced in a repository without opening the canister again to handle the bare fuel assemblies.

A range of overpack materials and types is available to suit different applications (SKB 2006; Andra 2005a; Andra 2005b; Andra 2005c; Andra 2005d; Andra 2005e; McKinley et al. 2006; Hardin et al. 2011a) and even different disposal environments if multiple repositories are used.

Other materials provide corrosion resistance under chemically-reducing conditions present in salt, clay, or shale host media. For example, the Swedish program has been developing waste packages made from pure copper, which has a very long lifetime in the KBS-3 concept planned for granite. Pure metals and metalliferous minerals can survive indefinitely under reducing conditions in nature, as demonstrated by natural analogues discussed in Appendix B (BSC 2004b).

3.4.1.2 Reprocessing Waste

As discussed in Section 3.2.3, reprocessing methods include aqueous processes such co-extraction, new-extraction, and UREX, as well as electro-chemical processes. Reprocessing extracts fuel value and also separates many radionuclides that can contribute to the radiological hazard associated with waste, for reuse or separate disposal. Heat generation by high-level radioactive waste decays rapidly because the greatest contributors are short-lived fission products (mainly isotopes of Cs and Sr, which are present in used fuel). Reprocessing may produce multiple waste forms each containing different categories of radionuclides from used fuel, potentially allowing tailored packaging and disposal (e.g., aqueous UREX process or electro-chemical processes). Because a range of dry and liquid wastes including low-level radioactive, mixed waste, and greater-than-Class-C waste are generally produced, it is appropriate to plan for different waste types and disposal requirements. Some reprocessing wastes are radioactive but do not produce much heat; these can be disposed of in openings within a geologic repository (Hardin et al. 2011a).

The borosilicate glass waste form remains the primary choice for use with aqueous reprocessing in the U.S., France, Japan, and the U.K. (Lutze 2006). With proper formulation and treatment the borosilicate glass waste form has a degradation rate under aqueous environmental conditions that is comparable to commercial Pyrex glass in terms of chemical durability (Lutze 2006).

Glass degradation can occur in a repository after the glass is exposed to moisture, and has been extensively studied (BSC 2004a). A recent investigation involving burial of radioactive high-level radioactive waste glass in soil under controlled conditions for 24 years, confirmed that the glass matrix degrades very slowly. Pu and other actinides were not released in this test, while Cs and Sr isotopes were released and transported by diffusion (Jantzen et al. 2008). Borosilicate glass should be maintained at less than approximately 500°C (transition temperature) to prevent crystallization and maintain a well-mixed state, and there are limits on solubility of molybdenum and other fission products that limit waste loading in the glass. However, these limits are readily met by established handling and storage methods (Carter and Luptak 2010) and can be readily accommodated by repository disposal concepts (Hardin et al. 2011a). Degradation performance of borosilicate glass and other glass waste forms continues to be actively investigated.

Developmental Waste Forms—Much research has been done, and is currently underway, to develop advanced waste forms, especially in association with development of new reprocessing methods. These waste forms are typically tailored to specific fractions separated from waste streams, exploiting chemically stable compositions and forms.

As an example, the UREX 1a process (Carter and Luptak 2010) separates alkaline/alkaline earth elements and produces a composite of Cs/Sr/Ba/Rb which includes the major heat-generating nuclides. The waste is treated by mixing with bentonite clay in a 25–75 mixture, then high-pressure pressing, and high-temperature sintering to produce ceramic pucks. These are loaded into canisters containing 120 kg of ceramic high-level radioactive waste.

Waste from advanced electro-chemical reprocessing (Goff and Simpson 2009) could include a metallic phase containing fission products such as transition metals and rare earths. Such products typically have low solubility and low rates of degradation in water, and may be disposed directly or mixed in other waste forms such as high-level radioactive waste glass.

The electro-chemical process demonstrated in Idaho uses zeolite to trap excess salt and fission products (Goff and Simpson 2009; Carter and Luptak 2010). Additional zeolite is added and the mixture is bonded with approximately 25% borosilicate glass. The glass bonded zeolite is cast into a 2-ft diameter by 15-ft tall canister containing 2,900 kg of glass. The waste form is 25% glass binder. The electro-chemical process also separates lanthanides which are converted to a lanthanide-based glass and cast into small diameter canisters.

Finally, advanced materials such as carbide ceramics are highly resistant to degradation in the disposal environment, and are being evaluated as waste forms as well as reactor fuels (Soelberg et al. 2010).

High-Level Radioactive Waste Packaging—High-level radioactive waste glass is generally poured into stainless steel canisters, which are typically closed by automated welding. Stainless steel is a relatively inexpensive material that resists degradation from elevated temperature during the initial pour, is readily formed and welded, provides structural support for handling and storage, and further resists corrosion prior to permanent disposal. Pour canisters have substantial strength for handling and storage functions, but are thin compared to overpack assemblies. Disposal overpacks can provide whatever additional strength or containment may be needed for a particular disposal concept (Hardin et al. 2011a). For some disposal concepts no overpack or further packaging may be needed for high-level radioactive waste, as proposed for the generic salt repository (Carter et al. 2011a) and for the French disposal concept (Andra 2005a; Andra 2005b; Andra 2005c; Andra 2005d; Andra 2005e).

Co-Disposal of Low-Level Waste—Low-level waste is typically packaged in sealed drums or other containers made of metal or concrete. For many disposal concepts there is enough space in the repository openings to dispose of low-level waste from reprocessing, repository operations, etc. (Hardin et al. 2011a), particularly greater-than-Class-C waste which has higher radioactivity than low-level waste typically disposed by near-surface burial.

3.4.2 Summary of the Engineered Barrier System Processes

The processes that affect the performance and safety functions of the engineered barrier system include coupled thermal, hydrologic, chemical, mechanical, biological, and radiological processes as well as other processes, many of which also occur in the natural system. However, additional processes not generally important in the natural system, such as material and structural degradation (e.g., corrosion and various chemical processes), are introduced into the system in conjunction with their corresponding man-made materials and environment. The primary processes relevant to the engineered barrier system are:

- **Thermal Processes**—Initiated by decay heat from the emplaced waste and affect the hydrologic, chemical, and mechanical environments in the engineered barrier system. Thermal processes include conduction, radiation, and convection and affect flow processes through evaporation, condensation, and buoyancy effects. They effect chemical processes such as mineral precipitation and dissolution, as well as characteristic properties of the porous host rock, such as thermal conductivity, relative permeability, and stress coefficients. The thermal effects on the mechanical environment are through thermal stresses and corresponding effects on engineered material strength.
- **Hydrologic Processes**—Include climate change, liquid- and gas-phase flow in the engineered barrier system and surrounding DRZ, flow diversion, capillarity, imbibition, evaporation, and condensation.

- **Chemical Processes**—Include a range of chemical processes that affect the degradation and transport mechanisms acting on or in engineered features. Examples of these processes are mineralogic alteration including phase changes and dehydration, dissolution and precipitation, metal oxidation and corrosion, gas phase generation, and colloid formation and stability.
- **Mechanical Processes**—Include a range of stress-strain processes that affect the degradation of engineered features and alter the host rock in the DRZ. These mechanical processes include rockfall, drift collapse, stress corrosion cracking of the waste containers, and creep deformation of the engineered components and backfill.
- **Biological Processes**—Include the potential effects of microorganisms on other processes relevant to performance, such as microbial effects on chemistry.
- **Radiological Processes**—Include the potential effects of ionizing radiation resulting from the decay of radioactive materials. An example is radiolysis in the pore water of the engineered materials, which can influence the concentration of chemical constituents in the water, which can in turn alter the rate of various material degradation processes.
- **Transport Processes**—Affect the rate of movement of released radionuclides and include such processes as advection, diffusion/dispersion, matrix diffusion, retardation, and colloid filtration.

An important aspect of these individual processes is their coupling within the energy, mass, and momentum conservation equations. The relative importance of these processes to overall repository system performance is a function of the specific disposal system. For example, for a repository system in bedded salt the host rock (geosphere) is a sufficient barrier by itself for the expected evolution scenario (i.e., no disruptive events), which makes the safety functions of the engineered components relatively less important, even though they must be present in order to fulfill the multiple barrier requirement. In contrast, for a disposal system with very robust waste forms and waste packages, geosphere processes are seen to be relatively less important, implying a reduced need for their detailed characterization. An example of this is the Swedish KBS 3 concept where copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500-m depth in saturated, granitic rock (SKB 2006b; SKB 2011).

The Joint EC/NEA Engineered Barrier System Project (Bennett 2010) compiled a "state-of-the-art" report on engineered barrier systems (Organisation for Economic Co-operation and Development 2003a), based on results from a questionnaire survey of waste management organizations and regulatory authorities, and their technical support organizations. This provided a snapshot of the status of the various disposal programs in 2003, reviewed the role of the engineered barriers in the different disposal concepts, identified the components of the engineered barrier systems, and documented their primary roles and functions. Based on this survey, the key processes that must be modeled in the engineered barrier system, in order to correctly predict repository performance, included all of those listed above to varying degrees depending on the host rock environment and the repository design (Organisation for Economic Co-operation and Development 2003a, Table 5.1). For a robust licensing safety case, safety assessment and process models that address these processes must be considered. For the generic safety case and models described here, some simplifying assumptions have been made. For example, some of the generic safety assessment models considered in Section 4 (e.g., salt, granite) make the assumption that the repository is isothermal. This and other simplifying assumptions are sufficient at early stages of safety case development when the basic structure of the safety case is being constructed.

Because they are most directly related to the effects of liquid-phase movement and its associated interaction with other material phases, which is the primary mechanism for the transfer of radionuclides to the biosphere, the processes in the engineered barrier system that most significantly affect the three key safety functions of isolation, containment, and limited and/or delayed releases are:

- **Hydraulic Flux—**An important function of clay-based buffer and backfill materials (in clay and granite repositories), and crushed salt backfill (in salt repositories) is low permeability to liquid flow. Low permeability can cause diversion of flowing groundwater (e.g., in the host rock) around waste packages, and substantially eliminate the possibility of advective flow in the engineered barrier system (e.g., in backfilled drifts). Without advective flow, the principal mode of radionuclide transport is molecular diffusion (attenuated by tortuosity, size and charge effects, and chemical retardation). Accordingly, advective releases from salt and clay repositories are considered to be insignificant, while diffusion may be the only important natural mechanism for transport of released radionuclides.
- **Material Degradation—**Many of the engineered components (e.g., waste packages and containers), as well as some of the waste matrices, are metals; and the degradation rates of metals and other waste materials (e.g., metallic, vitric, or ceramic) in a reduced oxidation state are sensitive to the redox state of the aqueous phase in contact with these materials. Thus, the nature of the redox processes in the engineered components is important to several safety functions and capabilities of the engineered barrier system.
- **Radionuclide Mobilization—**Mobilization of radionuclides in the aqueous phase, following degradation of engineered components, is sensitive to radionuclide solubility limits. Low solubility can substantially reduce the rates of release from the waste form, and subsequent transport through the engineered barriers. Some solubility limits depend on the local chemical conditions (e.g., whether oxidizing or reducing) while others are effective across a wide range of possible conditions. Solubility is more important in the engineered barriers, close to the waste form where concentrations are highest, compared to the natural system (far field).
- **Radionuclide Sorption—**Engineered barrier system components can significantly retard the transport of radionuclides released from the waste form, chiefly by sorption (e.g., ion exchange, surface complexation). This may occur as a primary function (sorption in clay buffers or radionuclide “getters”) or as a secondary function involving the products of metal corrosion.

These represent expected processes, i.e., those that are expected to occur through the natural temporal evolution of the system after repository closure. In some systems, e.g., salt and deep borehole disposal, these engineered barrier system processes are not expected to significantly influence the ability of the entire system to isolate and contain the waste because the natural barrier system is very robust. For those concepts, the most likely phenomenon for release of radionuclides to the biosphere is some sort of disruptive event and the associated physical and chemical release processes. This would include tectonic processes such as volcanism or seismicity and human-induced processes such as inadvertent or deliberate intrusion into the repository caused by nearby mineral exploration. For example, in the case of the WIPP repository, the most important processes might be drilling intrusion into the repository and subsequent communication by brine movement through the drill hole to an aquifer. For any repository concept, the key processes in the engineered (and natural) system, whether expected or disruptive, will be elucidated by a comprehensive FEP analysis and scenario development process, as described in Section 4.2.

3.5 External Events Acting on Natural and Engineered Barriers

External events have the potential to influence the barrier capability of the natural and engineered barrier systems. For generic disposal in suitable salt, clay or deep borehole, the only likely release and migration of radionuclides to the biosphere is associated with external events. These disposal options will be very robust under undisturbed conditions. Because of the great depths involved, external events are expected to have little impact on disposal in deep boreholes. A mined repository would be deep enough below the present land surface to ensure that the waste is not exposed to the biosphere through erosion or denudation during its hazardous period.

Climate Changes—The radioactivity of nuclear waste will decay over a period of time (100,000 years or longer) in which major environmental changes are expected. Climate change can impact the surface and subsurface environments through changes in a variety of factors such as precipitation, glaciation, permafrost, sea level, erosion and deposition, infiltration, and soil and bedrock hydrologic properties. The flow and transport patterns may be altered by changes in regional recharge of the natural system as well as by changes in the elevation of a water table. Estimates of potential impacts from climate change are highly site specific. Although equally applicable to repositories located in generic salt, granite, and clay locations and to a lesser extent to deep borehole disposal, the following example of potential climatic impact on barrier performance is offered with respect to glaciation and granite.

Application of the KBS-3 disposal concept in northern latitudes where Pleistocene glaciation occurred presents the likelihood that a glacial climate will return during the repository performance period. Return of continental glaciers is part of the expected future in the SR-Can assessment (SKB 2006b). The potential impacts on the natural system can include:

- Decreased temperature and the possibility of freezing
- Larger hydraulic gradient, increased flow velocity
- Deep penetration of melt water
- Reduction in salinity
- Increased faulting and fracturing caused by the increase in groundwater pore pressure and the isostatic response of the lithosphere to glacial loading

The hydraulic effects from a kilometer or more of glacial surface relief would affect flow conditions and the chemistry of groundwater; however, the residual effects on the hydraulic structure and mineralization of the aquifers likely would be limited to present-day conditions (which have already resulted from more than two million years of recurring glacial exposure) (SKB 2006b).

Faulting will likely occur along existing discontinuities, especially those which responded to previous glacial loading and unloading. These discontinuities would be mapped and avoided during repository development. Some new faulting would be expected in response to glaciation; however, the displacements likely to occur on new faults would likely be much smaller than on existing faults. Given the buffer thickness, displacements of less than 10 cm are not expected to cause canister failure or significantly disrupt buffer function. The likelihood of new faults with greater than 10 cm displacement, intersecting vertically emplaced waste packages, would likely be insignificant (SKB 2006b). If this were not the case buffer thickness would need to be re-engineered to accommodate likely fault displacements.

Climatic events have little impact on the barrier capability of deep borehole disposal. Little hydrologic communication exists between near surface groundwater and waters found at depth. At depths in excess of 3 km there is very little likelihood of erosional or other climate related processes impacting deep borehole disposal.

Seismic Events—Depending on magnitude, seismic events have the potential to damage engineered components and alter flow in the natural system. A site located in a suitable generic salt, clay, or granite formation including a deep borehole would have a low probability of seismicity that could damage the engineered barriers. In a granite or crystalline rock formation the strength and integrity of the host rock provides superior protection of the waste packages from rock pressures over long periods of time and would greatly contribute to the long-term isolation of radionuclides from the biosphere. Similarly, a deep borehole disposal location could be sited to have a low probability of seismic events that could damage natural or engineered barriers. Because of the effectiveness of the hydrologic barrier functions of the natural system associated with deep borehole disposal, release of radionuclides resulting from seismic activity would be limited in extent and not enter the biosphere. In the case of salt, the plasticity of the host salt will lead to creep closure around the waste packages, which confines their movement and protects them from potential seismic impacts. In a generic repository in clay, the buffer material will provide some protection to the waste packages.

Igneous Events—Direct release of radionuclides to the biosphere could occur if a magmatic conduit for a volcanic eruption intersected the waste disposal zone. The presence of igneous rocks of Quaternary age at the surface or intersected by the borehole would indicate a potentially significant probability of future igneous activity and associated impacts on repository performance. These areas should be avoided.

Human Intrusion—Hydro-carbon resources can be located in the vicinity of salt and clay/shale formations, and water resources may be present in or near many rock types. Exploratory drilling that intersects or is close to the repository could bypass or compromise the natural barrier system.

For deep borehole disposal, the depth of the waste disposal zone and the 3-km borehole seal zone would restrict or prevent access to the waste. The site location could be chosen such that there are no known significant natural resources or geothermal heat sources nearby that might encourage exploratory drilling. The groundwater at depth would likely be of limited use (too salty) and uneconomical to exploit. The borehole would be positioned within a region of rock with low permeability, which would be inconsistent with groundwater resource use. The presence of ore deposits would be difficult to assess from the surface in deeply buried crystalline basement rocks and if resource were there, they may well be uneconomical to develop. Because of the vertical disposal of waste in a deep borehole, the footprint for inadvertent intrusion is small. This further reduces the likelihood of an inadvertent intrusion into the waste disposal zone.

A repository in a granite formation would likely have a low probability of human intrusion. The depth of the repository would restrict or prevent access. The site location could be chosen such that there are no known significant natural resources or geothermal heat sources nearby that might encourage exploratory drilling. For example, a location where the groundwater at repository depth is of limited use (too salty) and uneconomical to exploit. Moreover, the repository would be positioned within a region of rock with low permeability, which would be inconsistent with groundwater resource use. For these reasons, it is unlikely that wells would be drilled into the repository in search of potable water or other resources.

Both salt and clay/shale formations are unlikely targets for future drilling to recover potable water. However, sedimentary rocks can provide a more attractive target for hydrocarbon exploration than other formations. Measures to reduce the future likelihood of human intrusion, for example, by avoiding areas with proven resources, may be appropriate for repositories sited in salt or clay/shale formations, and safety assessments can take into account the potential consequences of intrusion. For the WIPP, safety assessments demonstrated that even with the assumed probability of future drilling intrusion close to one, performance was well within regulatory limits (U.S. Department of Energy 1996a; U.S. Department of Energy 2004a).

3.6 Summary of Previous Safety Assessments

The United States and other countries around the world have conducted numerous detailed safety assessments evaluating a variety of geologic disposal options, including those relevant to the four options being considered in the Generic Safety Case. Built upon extensive laboratory and field research, these assessments provide insight into potential repository behavior and the complementary role of engineered and natural barriers. Examples of these safety assessments are summarized below. The first group consists of three assessments conducted in the United States to support historical policy and regulation development. The second group consists of previous assessments undertaken by the United States and other countries for applications ranging from generic scoping studies to safety case development. Appendix C discusses these two groups in more detail. The results of the generic postclosure safety assessments presented in Section 4 are generally consistent with the results of the safety assessments described below.

Historical Studies Supporting Policy Development in the United States—The three studies described below were conducted to support the historical development of policies and regulations in the United States for nuclear waste disposal. An important aspect of these studies was the identification of criteria that could be important to assessing repository safety.

- **Environmental Impact Statement on the Management and Disposal of Commercially Generated Radioactive Wastes—**For this study, the repository conceptual design consisted of emplacing waste packages in boreholes in a mined repository, which would then be backfilled (U.S. Department of Energy 1980). The study focused on four rock types: salt deposits (bedded and domal), granite, shale, and basalt. Of the factors considered relevant to geologic disposal, the most important ones were thought to be the hydrologic regime; the tectonic regime; the multi-barrier concept; and thermal, physical, and geochemical properties of the host rock. In addition, criteria were identified to use for site selection to ensure that the natural barrier functioned as planned. Based on the study results, the Environmental Impact Statement concluded that a nuclear waste repository could be sited, loaded, and sealed with every expectation that long-term radiological impacts would be nonexistent.
- **Waste Isolation Systems Panel**—A panel of the Board on Radioactive Waste Management of the National Research Council conducted safety assessments to investigate disposal of high-level radioactive waste, transuranic waste, and other special long-lived wastes (National Academy of Sciences 1983). The potential host rock types considered included salt (bedded and domal), granite, basalt, rhyolite (tuff), unsaturated alluvium over a regional aquifer, and granite/metamorphic rocks under a regional sedimentary aquifer. For evaluation purposes, the panel developed performance criteria based on the annual or lifetime radiation dose to an individual exposed at some future time to radionuclides released to the environment from a geologic repository. The study revealed two key system performance elements: (1) the absence of flowing groundwater that in time could come in contact with the waste form, and (2) low, solubility-limited release rates of the radionuclide products in the waste form, geologic retardation, and a decrease in potential radiation doses to individuals resulting from the dispersion and dilution processes during transport and discharge in surface water.
- **Background Information Document for the U.S. Environmental Protection Agency Rule 40 CFR Part 191**—In this report, the U.S. Environmental Protection Agency presented risk assessments to support development of the regulatory standard in 40 CFR Part 191 for high-level waste and transuranic waste repositories (U.S. Environmental Protection Agency 1985). The geologic settings included specific sites in bedded salt, basalt, and unsaturated volcanic tuff as well as sites for granite that were either more regional in nature or idealized. The various sites were evaluated with a standard set of engineering assumptions because it was determined that the models were not highly sensitive to these assumptions. The model results provided consequence estimates in terms of population risk rather than individual risk.

Previous Safety Assessments Relevant to the Four Geologic Disposal Options—The United States and other countries around the world have conducted numerous safety assessments over the years. The following provides a sampling of the safety assessments, organized according to relevance to the four disposal options.

- **Salt**
  - *Review of Salt Repository Science (United States)*—In 2011, Sandia National Laboratories published a high-level review of salt repository science and proposed a framework for evaluating a generic salt repository (Hansen and Leigh 2011). The report also presented a gap analysis identifying areas of remaining uncertainty in important FEPs. Besides recognizing the positive qualities of salt for waste isolation, the study concluded that (1) the United States has an abundance of salt formations of sufficient depth, thickness, and lateral extent to accommodate a repository, (2) experience gained from WIPP as well as the programs of other countries offers

substantial benefit, and (3) the knowledge base regarding salt is likely to be expanded by advancements in multi-physics modeling and laboratory testing.

- *Waste Isolation Pilot Plant Recertification (United States)*—Operated by the U.S. Department of Energy, WIPP disposes defense-generated transuranic wastes in thick salt beds located in New Mexico (U.S. Department of Energy 2009). Since 1996, the U.S. Department of Energy has conducted five different safety assessments. The results from the latest safety assessment in 2009 as well as the previous assessments and continuing scientific studies have shown that WIPP is operating and performing as expected. The dominant factors affecting the undisturbed system behavior are fluid flow, rock deformation surrounding the excavation, and waste degradation. While WIPP disposes of transuranic waste, the results can still provide useful insights regarding the potential behavior of a salt repository used for the disposal of high-activity waste.

- *Gorleben Preliminary Safety Assessment (Germany)*—Germany has been investigating salt as a possible host rock since the 1960s (Weber et al. 2011). Efforts on a preliminary safety assessment began in 2010 using the salt dome at Gorleben as a reference site. The safety concept emphasizes the systematic demonstration of long-term safe confinement of high-level radioactive waste by demonstrating the long-term effectiveness and integrity of the geological and engineered barriers. The domal rock salt has multiple characteristics contributing to barrier performance including being generally dry, virtually impermeable, and self-healing. Seals and backfill are being investigated as engineered barriers. The results thus far indicate that the reference site is expected to maintain its integrity as a geologic barrier during the entire reference period of 1,000,000 years.

- **Clay**

  - *Generic Clay Study (United States)*—Sandia National Laboratories conducted scoping performance analyses as part of a recent feasibility study on high-activity waste disposal in clay formations (i.e., mudstone, claystone, shale, and argillite) within the United States (Hansen et al. 2010). The study found that clay mineralogy and chemistry combine to limit radionuclide transport through the influence of low permeability, chemically-reducing environments, and sorption. Multi-physics calculations suggested that the maximum extent of radionuclide transport will be on the order of tens to hundreds of meters, or less, in 1,000,000 years. Because of the long transport times, most of the mobile radionuclides will decay before they can reach the biosphere.

  - *Opalinus Clay Safety Case (Switzerland)*—After years of research, the National Cooperative for the Disposal of Radioactive Waste (Nagra) in 2002 presented its safety case for waste disposal in the Opalinus Clay in Switzerland (Nagra 2002). To better understand how the natural and engineered barriers work together, Nagra analyzed over 20 cases run as part of the Reference Scenario and 49 cases run to evaluate other scenarios and circumstances. The associated results were below the applicable regulatory guideline. In the reference case, most radionuclides decay within the waste forms and buffer material. Because transport is dominated by diffusion, only the most mobile and long-lived radionuclides reach the edge of the host rock formation. The clay-rich confining units surrounding the host rock have good sorption properties and can also retard transport. Long-lived radionuclides that escape the confining units may enter regional aquifers, where they are dispersed and diluted. Additional dilution occurs when the deep aquifers discharge to the more dynamic freshwater flow systems of near-surface gravel aquifers or to river waters.

  - *Dossier 2005 for Clay (France)*—In 2005, the French National Radioactive Waste Management Agency (Andra) submitted Dossier 2005 Argile to French authorities after nearly 15 years of research on clay formations (Andra 2005a). The safety assessment and supporting scientific investigations demonstrate that the Callovo-Oxfordian argillites offer a robust and efficient means of disposal. The study identified three barriers that act to prevent or delay the release and transport of radionuclides: (1) the waste packages (primarily the ability to maintain integrity), (2) the disposal cells (diffusive transport), and (3) the geologic medium (immobilization due to smectite, favorable geochemistry causing precipitation, and diffusive transport). The assessment results found that there is no significant impact to man and the environment for the normal evolution scenario. In the altered evolution scenarios, Andra explored the potential impacts of low probability events and processes such as seal and plug failure, defective containers, and borehole penetration of the repository. The expected dose for each scenario remained well below the applicable regulatory limit. Even given the extreme situation in which all safety functions were assumed to be severely degraded, the expected dose was still less than the limit.

- *SAFIR 2 (Belgium)*—Research on long-term waste management began in Belgium as early as 1974. In 2001, *Safety Assessment and Feasibility Interim Report 2* (SAFIR 2) was released (ONDRAF/NIRAS 2001). The reference host formation in SAFIR 2 is the Boom Clay. Results of both normal and altered evolution scenarios confirm that, while the waste packages and backfill materials are expected to provide some barrier capability, it is the Boom Clay that dominates in terms of preventing or delaying radionuclide transport. Migration through the Boom Clay occurs primarily through diffusion. Sorption by the clay minerals or by organic materials in the clay will halt the transport of many radionuclides. The few radionuclides that migrate through the clay layer will be diluted in the overlying aquifers, where advection and dispersion dominate. Some sorption is also possible because of minerals present in the aquifers. The calculations for the normal evolution scenario indicate that the exposure for an individual in the reference group is below the applicable dose limit. The initial altered evolution results indicate that the overall system performance remained broadly intact despite the various assumptions compromising system integrity.

- **Granite**
  - *Generic* Granite *Study (United States)*—In 2011, Sandia National Laboratories conducted a study to evaluate the feasibility of using granite as the disposal host rock (Mariner et al. 2011). For scoping purposes, a preliminary safety assessment was conducted to analyze two scenarios: the defective waste package scenario and the buffer failure scenario. Results indicated that the most important simulated processes for preventing release are canister corrosion, waste form degradation, and radionuclide precipitation. These processes, in turn, depend highly on reducing conditions and the presence and properties of the canister and buffer. The buffer, in addition to delaying waste package failure, presents a diffusive and sorptive barrier to radionuclide transport; however, the results suggest that the buffer's role in limiting canister corrosion rates is more important to radionuclide release to the geosphere. Once radionuclides enter the geosphere, fracture flow velocities, matrix diffusion, sorption, and radioactive decay are of the highest importance to the dose rate at the receptor well. The importance of radioactive decay is magnified by the long residence times in the repository and geosphere compared to the relatively short half-lives of many of the radionuclides.
  - *Forsmark License Application (Sweden)*—The Swedish have researched deep geological disposal for more than 30 years (Thegerström and Olsson 2011). Forsmark was selected as the repository site in 2009. The safety assessment, SR-Site, was conducted as part of the license application (SKB 2011; Hedin et al. 2011). The SR-Site results confirmed the favorable properties of the Forsmark site. The crystalline bedrock at Forsmark has relatively few open or partly open fractures at repository depth. Diffusion is the dominant transport mechanism in the rock matrix, which also has favorable sorption properties. A reducing chemical environment and salinity at repository depth help ensure the stability of the bentonite buffer. The results also point to the waste package as an important barrier; canister failures are expected to be rare. Even in scenario

analyses involving various modes of canister failure, the radiological consequences remain below the applicable regulatory limits.

- *Posiva's Preliminary Safety Case for the Olkiluoto Site (Finland)*—The Olkiluoto site in Finland has been studied for over 20 years (Posiva 2010a). Posiva has conducted safety assessments on Olkiluoto to support license application development. The results indicate that the key barriers are the canister, the bentonite buffer, the backfill (specific to one of the design variants), and the host rock. Most of the canisters are expected to remain intact over at least several hundreds of thousands of years. Once a canister fails, the following barrier features contribute to performance: (1) low groundwater flow rates, (2) low dissolution rates of spent fuel under reducing conditions and low corrosion rates of fuel assembly materials, (3) low solubilities of several of the most hazardous radionuclides, (4) slow transport of radionuclides through the bentonite buffer, and (5) slow transport of radionuclides through the host rock.
- *Third and Fourth Case Studies in Granite (Canada)*—Canada has been investigating various disposal strategies for about 30 years (Gierszewski et al. 2004; Kremer et al. 2011). The Third and Fourth Case Studies, presented in 2004 and 2011 respectively, considered hypothetical repositories representative of potential sites that could exist within the crystalline rock in the Canadian Shield. The safety assessments showed that, while robust containers are the primary barriers to radionuclide release, the other engineered and natural barriers are effective in preventing or delaying radionuclide transport. Even in sensitivity cases in which all the containers were assumed to fail, the calculated dose rates to the critical group stayed below the applicable dose rate constraint and the natural background radiation. This result illustrates the benefits of a multiple barrier system in which the engineered sealing materials and the geosphere are capable of contributing to barrier performance.

- **Deep Borehole**
  - *Generic Deep Borehole Study (United States)*—Although the concept of the deep borehole option has been investigated off and on since the 1950s, it has been re-examined recently by Sandia National Laboratories (Brady et al. 2009; Swift et al. 2011). A preliminary safety assessment was conducted to study transport in the borehole as well as in the disturbed zone around the borehole. The results indicate that radionuclides, once emplaced, tend to stay in the borehole or its immediate vicinity. The fuel and the majority of the radionuclides in it will be thermodynamically stable and will resist dissolution into borehole fluids. Thermally driven flow causes upward advective transport for only about 200 years. In the subsequent ambient conditions, there is only diffusive transport. Over the 1,000,000-year period, diffusive transport cannot move radionuclides through the various media more than about 200 m. On this timescale, the vast majority of radionuclides will have decayed.

4 GENERIC POSTCLOSURE SAFETY ASSESSMENTS

4.1 Introduction

This section describes a preliminary implementation of the safety assessment methodology leading to quantitative evaluations of the postclosure safety of four generic geologic disposal options: mined disposal facilities in salt, clay, and granite and deep borehole disposal in crystalline rock.

Generic scenario analysis is described in Section 4.2. The scenario analysis includes FEP identification and screening following by scenario construction and screening. Section 4.3 describes the development of safety assessment models for each of the four generic disposal options that describe selected FEPs and scenarios. This involves numerical representations of the selected disposal system scenarios for producing quantitative estimates of disposal system performance and safety. The generic safety assessment model results for each of the geologic disposal options are presented in Section 4.4. These simulation results include both deterministic analyses and sensitivity analyses.

4.2 Scenarios

4.2.1 Introduction

Within the safety assessment methodology in Section 2.3.1, scenario analysis is captured in Step 3:

3. Identify scenarios for analysis
   a. Identify and screen potentially relevant features, events, and processes (FEPs)
   b. Construct and screen scenarios
   c. Estimate scenario probabilities

Scenario analysis, also referred to as scenario selection, is commonly used to support postclosure safety assessments (Organisation for Economic Co-operation and Development 1999a, Section 1). Early development of scenario analysis methodologies was performed by the U.S. Nuclear Regulatory Commission (Cranwell et al. 1990) and the Nuclear Energy Agency (Organisation for Economic Co-operation and Development 1992; Organisation for Economic Co-operation and Development 1999a). More recent discussions of scenario analysis methodologies are provided in a number of reports (U.S. Nuclear Regulatory Commission 2003, Section 2.2.1.2; BSC 2005b; Organisation for Economic Co-operation and Development 2006; Sandia National Laboratories 2008a). Common to these methodologies are the following scenario analysis steps:

- **FEP Analysis—**Identify, classify, and screen FEPs potentially relevant to the long-term performance of the disposal system
- **Scenario Development—**Construct and screen scenarios from the screened in (included) FEPs
- **Model Implementation—**Specify the numerical implementation of the screened in scenarios in the safety assessment model

There are a number of definitions of scenarios and features, events, and processes in the literature. A good and consistent set of definitions is (U.S. Nuclear Regulatory Commission 2003, Section 3):

- A *scenario* is a well-defined, connected sequence of features, events, and processes that can be thought of as an outline of a possible future condition of the repository system. Scenarios can be undisturbed, in which case the performance would be the expected, or nominal, behavior for the system. Scenarios can also be disturbed, if altered by disruptive events.
- A *feature* is an object, structure, or condition that has a potential to affect disposal system performance. The waste package is an example of a feature.

- An *event* is a natural or human-caused phenomenon that has a potential to affect disposal system performance and that occurs during an interval that is short compared with the period of performance.
- A *process* is a natural or human-caused phenomenon that has a potential to affect disposal system performance and that operates during all or a significant part of the period of performance.

A FEP generally encompasses a single phenomenon; typically it is a process or event acting upon or within a feature (or region).

Scenario analysis has two primary purposes in the development and documentation of a postclosure safety assessment. First, scenario analysis supports the demonstration of the comprehensiveness of the FEPs and scenarios - it can provide objective evidence that all potentially relevant FEPs have been identified and that all reasonably possible future states of the disposal system have been considered. Second, scenario analysis supports the demonstration of completeness of the safety assessment model – it can provide a structure to ensure that all important FEPs, scenarios, and their uncertainties are quantitatively represented in the model. These objectives are supported by formal documentation of scenario analysis process including the treatment of included FEPs (how they are included in the model) and excluded FEPs (the rationale for exclusion).

FEP analysis, scenario development, and safety assessment model development and implementation are iterative processes that are dependent on site-specific information, design alternatives, regulations, and performance metrics and evolve as new information (e.g., experimental data, model results, socio-political drivers) becomes available.

Scenario analysis is typically conducted for a specific site and disposal concept. Applying scenario analysis to a generic geologic disposal system requires a number of assumptions to be made because regulatory requirements, site characterization data, and design information may not be available. Additionally, identification and evaluation of the external factors (i.e., disruptive events) that influence scenario development is, for the most part, site specific. To support this Generic Safety Case, these assumptions derive from the safety strategy in Section 2 and the assessment basis in Section 3.

In the early stages of a disposal program, prior to the selection of disposal options or a site, this type of generic scenario analysis is sufficient. The basis for generic FEP analysis and scenario development is subjective in nature because sufficient design- and site specific information is not available. Success is improved by drawing on existing and accepted FEP lists from other countries as well as previous performance and safety assessments for a wide range of geologic media. Simplified disposal system models may be constructed and analyses may be used to inform research needs for the next iteration as well as to identify the strong points of the disposal options under consideration.

As the disposal program progresses through various phases, the available information becomes more site specific and the FEPs and scenarios can be refined, making use of the new information collected from the research and development program and updated models. This process is repeated until sufficient information and pedigree exists to support a license application.

An overview of FEP analysis and its application to the generic safety case is presented in Section 4.2.2. Parallel information for scenario development is presented in Section 4.2.3.

4.2.2 FEP Analysis

FEP analysis consists of the following steps:

1. **FEP Identification**—Identify and classify FEPs potentially relevant to the long-term performance of the disposal system.
2. **FEP Screening**—Screen the FEPs using specified criteria (e.g., low probability, low consequence, by regulation) to identify those FEPs that may be excluded from a safety assessment model and those that should be included in the model.

A summary of FEP identification and screening approaches and considerations is presented in Section 4.2.2.1. The application of FEP identification to support the generic safety case is described in Section 4.2.2.2. Preliminary FEP screening for each of the four safety case disposal options is described in Section 4.2.2.3.

4.2.2.1 Overview

Overviews of FEP analysis approaches are presented in Organisation for Economic Co-operation and Development (1992, Section 4), BSC (2005b, Section 2), and Freeze et al. (2010, Section 1). Summary information is provided in Section 4.2.2.1.1 for FEP identification and Section 4.2.2.1.2 for FEP screening.

4.2.2.1.1 FEP Identification

The objective of FEP identification is to identify and categorize a comprehensive list of FEPs potentially relevant to the long-term performance of the disposal system. Potentially relevant FEPs are all phenomena (i.e., combinations of process and events acting upon features) that might possibly influence the long-term performance of the disposal system, regardless of importance (e.g., no matter how improbable or inconsequential). Formal FEP identification, which includes FEP categorization, can be used to support the demonstration of the comprehensiveness of the FEP list by providing objective evidence that all potentially relevant FEPs have been identified – it can address the question, "Have we thought of everything?" (Freeze et al. 2010, Section 1.1.1).

FEP identification can be accomplished using either top-down or bottom-up approaches. Top-down approaches tend to be more systematic, deriving from a small number of broadly-defined endpoint performance metrics. A top-down system is expanded by identifying influences on each metric, such that there are increasingly larger numbers of more detailed phenomena as each successive level of influence is developed. Bottom-up approaches tend to be more free-form, deriving from a large number of more narrowly-defined independent phenomena that describe the system being considered. Initial FEP identification is most commonly performed using bottom-up approaches such as development from existing detailed FEP lists and/or brainstorming (i.e., freely-structured identification of FEPs) by groups of relevant experts (Organisation for Economic Co-operation and Development 1999a). Numerous comprehensive FEP lists are available in the literature to serve as a starting point, including the International FEP Database (Organisation for Economic Co-operation and Development 1999a; Organisation for Economic Co-operation and Development 2006) and FEPs documented in Sandia National Laboratories (2008a; 2008b).

FEP lists are commonly classified in accordance with top-down-structured and/or hierarchical categorization schemes. The cross-mapping of a bottom-up-generated FEP list with top-down-structured categorization schemes helps to uncover missing FEPs and interactions and can provide a framework for demonstrating comprehensiveness. The following statements provide general observations regarding FEP categorization:

> *By classifying features, events, and processes under different schemes, information on additional phenomena and interaction can be gained. … classification schemes that examine the system from different viewpoints should be used.* (Organisation for Economic Co-operation and Development 1992)
>
> *…it is helpful to have a structure or categories so that the completeness (of categories and within categories) can be assessed, and equivalent levels of detail guided, e.g., similar numbers of FEPs might be found in each category* (Organisation for Economic Co-operation and Development 1999a)
>
> *Confidence in the comprehensiveness of the list of factors is developed by organizing and ordering the information in many different ways.* (Goodwin et al. 1994)

> *The [International FEP Database] classification scheme captures a range of radioactive waste disposal assessment projects within its scope. …this will be an aid to achieving comprehensiveness of assessments …* (Organisation for Economic Co-operation and Development 1999a)

An important consideration in the development of a FEP list is the level of detail at which the FEPs are defined. There is no uniquely correct level of detail at which to define and/or aggregate FEPs. Too little detail produces a short list of broadly defined FEPs, where it becomes difficult to isolate important issues for screening or implementation in a model. Too much detail produces a long list of narrowly defined FEPs, where it becomes impractical to develop independent screening decisions for each FEP. The following observations relate to identifying a comprehensive list of FEPs (Organisation for Economic Co-operation and Development 1999a):

> *A list that is too general will not be useful. On the other hand, a list that descends to a too-detailed level ... will tend to become incomplete as it becomes more difficult to be comprehensive at more detailed levels.*
>
> *It is impossible to demonstrate comprehensiveness or completeness, in the sense that it is impossible to exhaustively identify all possible FEPs and interactions within a complex and evolving system. It is possible, however, to list a range of broadly-defined FEPs that might be relevant to consider in safety assessments. This is the aim of the International FEP List: to be comprehensive in a broad sense rather than in a detailed sense.*

Regardless of the level of detail explicitly captured in a FEP, it is common that it can subsume, either explicitly or implicitly, several very specific issues and/or finer details, all of which are more broadly addressed by the FEP and its associated screening evaluation. However, no matter what level of detail is provided in a FEP list, the "next finest" level of detail is often desirable to better support the specific implementation in a safety assessment model. Therefore, a balance must be struck between a broader level of detail desirable for screening (and to support a demonstration of comprehensiveness in a broad sense) and a finer level of detail desirable for the safety assessment model implementation. Also, the level of detail of FEPs need not be consistent across the entire disposal system. It is common for there to be a greater number of more-detailed FEPs in the areas of the disposal system where more complex processes predominate. In practice, lists that aggregate phenomena at relatively coarse levels have proven to be suitable for evaluation in regulatory settings in the U.S. (e.g., the WIPP Compliance Applications (U.S. Department of Energy 1996a; U.S. Department of Energy 2004a) and U.S. Department of Energy 2008b; Sandia National Laboratories 2008a; and Sandia National Laboratories 2008b).

Ultimately, the comprehensiveness of a FEP list cannot be proven with absolute certainty. However, confidence can be gained through a combination of cross-mapping with multiple categorization schemes, audits/comparisons with other FEP lists, and formal and systematic reviews (Freeze et al. 2010, Section 1.1.1).

4.2.2.1.2 FEP Screening

The objective of FEP screening is to identify those important FEPs that should be included in the safety assessment model and those that can be excluded from further consideration. The included (screened in) FEPs represent a subset of the complete set of potentially relevant FEPs. FEP screening is typically performed by comparing FEPs with specified exclusion criteria. Common exclusion criteria include:

- **Probability—**FEPs having less than a specified probability of occurrence may be excluded from consideration in the safety assessment model on the basis of low probability. The low probability criterion will typically be specified in the regulations (e.g., exclusion of FEPs having a probability of occurrence of less than one chance in 100,000,000 per year).

- **Consequence—**FEPs that have little or no effect on disposal system performance, as defined by specified metrics, may be excluded from consideration in the safety assessment model on the basis of low consequence. The low consequence criterion will typically be specified in the regulations, although the consequence threshold may not necessarily be quantitative (e.g., exclusion of FEPs whose omission would not significantly change radionuclide exposure to a receptor or radionuclide releases to the accessible environment).
- **Regulation—**Some FEPs may be specifically excluded by regulations that limit the scope of analysis to specific characteristics, concepts, and definitions. FEPs related to the biosphere, disruptive events (e.g., human intrusion), and long-term geologic processes are commonly addressed through regulations.

Additional considerations in applying these exclusion criteria are discussed in BSC (2005b, Section 4.1).

The screening process is site-, design-, and regulation-specific. FEPs are considered one by one and are checked for interactions. A FEP need only satisfy one of the exclusion screening criteria to be excluded from the safety assessment model. A FEP that does not satisfy any of the exclusion screening criteria must be included (screened in) in the model.

During the FEP screening process, the selection of appropriate performance metrics is important. System-level performance metrics may include human health effects in the biosphere (e.g., annual dose); cumulative radionuclide release to the biosphere; or radionuclide concentrations in the biosphere. Subsystem-level performance metrics may include radionuclide mass flux and cumulative release across intermediate domain boundaries; radionuclide concentrations within domains; radionuclide mass in place (within domains and remaining in the waste form); engineered component degradation rates; and spatial distributions of various physical and chemical properties (e.g., pH, temperature, fluid saturation, chemical concentrations, dissolution and precipitation rates). Focusing only on system-level performance metrics may cause the relative performance contributions made by various subsystems to be overlooked. Ultimately, the assessment of importance depends on the type of decision that is being made about the disposal system (e.g., a comparison with a standard may focus more on system-level performance, whereas a comparison of specific design options such as different waste form types may focus more on subsystem-level performance).

Another consideration in FEP screening is the relative performance contributions of specific design components (e.g., features) or physical domains. For example, given a system-level performance metric for a repository system in bedded salt, the host rock is likely to be the most important feature of the disposal system, and the engineered components may be relatively unimportant, even though they may be effective. Similarly, in a disposal system with very robust waste forms and waste packages, sorption in the host rock is likely to be relatively unimportant, even though it may be effective.

The FEP screening decisions and supporting technical bases, including assumptions, should be documented in a clear and organized manner. As with FEP identification, FEP screening is an iterative process; new information, new designs, and new safety assessment results can result in revisions to FEP screening decisions.

4.2.2.2 FEP Identification for the Generic Safety Case

The Used Fuel Disposition Campaign has developed a list of potentially relevant FEPs for long-term disposal of used nuclear fuel and high-level radioactive waste for a range of disposal options (Freeze et al. 2010; Freeze et al. 2011). The development of these UFD FEPs, which are potentially relevant to the four generic safety case disposal options, considered existing FEPs from the following sources:

- The International FEP Database (Organisation for Economic Co-operation and Development 1999a; Organisation for Economic Co-operation and Development 2006). These FEPs are from 10 different national radioactive waste disposal programs covering a wide range of disposal system concepts and geologic settings.

- The FEP List presented in Sandia National Laboratories (2008b, Section 6.2). These FEPs are specific to a single disposal concept and geologic setting.

These existing FEPs were grouped and classified so that FEPs of similar or related scope could be consolidated, generalized, and given a more consistent level of detail (Freeze et al. 2001, Sections 2 and 3; BSC 2005b, Section 3; Freeze et al. 2010, Section 2). The resulting set of 208 UFD FEPs is listed in Appendix D, Table D-2. Each UFD FEP is defined by a “Description” at a broad level of detail such that it is potentially applicable to all of the disposal options. For example, “Sorption of Dissolved Radionuclides in the EBS” is potentially relevant to all disposal options. Each UFD FEP is further defined by additional details under “Associated Processes”. The level of detail captured by the FEP Descriptions and Associated Processes is appropriate for supporting a generic safety case. The technical scope of the 208 UFD FEPs collectively captures the full range of potentially relevant phenomena (and associated time- and length-scales) encompassed by the source FEPs (from the International FEP Database and from Sandia National Laboratories (2008b)) and is consistent with the generic natural and engineered features (Sections 2.2.2 and 2.2.3) and processes (Sections 3.3 and 3..4.2) that represent the four safety case disposal options. To provide traceability, the related FEPs from Sandia National Laboratories (2008b) are mapped to each UFD FEP in the table. The FEPs from Sandia National Laboratories (2008b) are in turn mapped to the International FEP Database (see Sandia National Laboratories 2008b, Appendix F). Because the UFD FEP list derived from existing FEP lists, it benefits from the extensive reviews and the cross-mapping that were conducted to support the comprehensiveness of the predecessor lists.

The UFD FEP list is organized by a UFD classification scheme that is similar to the International FEP Database categorization scheme (Organisation for Economic Co-operation and Development 1999a, Section 3). In Figure 4-1, the UFD classification scheme and numbering hierarchy is shown together with the generic disposal system domains (features). The generic features and domains in Figure 4-1 are the same generic features and domains introduced previously in Figure 2-1. The numbers associated with various domains, features, events, and processes in Figure 4-1 correspond to the FEP numbering system. Across the disposal system domains there is a consistent structure and numbering scheme for the features (2.x.01 contains the first feature, 2.x.02 contains the second feature, etc.) and the processes (2.x.07 contains mechanical processes, 2.x.08 contains hydrologic processes, etc.). The full FEP numbering hierarchy is shown in Appendix D, Table D-1.

Figure 4-1 also illustrates how each of the generic features can be acted upon by events (i.e., External Factors) and/or coupled multi-physics processes and indicates FEP categories that control the safety assessment model calculations (i.e., Assessment Basis and Radionuclide Exposure).

The generic UFD FEP list is easily expanded to capture additional details associated with specific repository designs and site geology as disposal programs mature. Most commonly, this expanded level of detail would be captured in the “Associated Processes” of an existing FEP. However, a new FEP could be created if a new or unique features or phenomena were identified as part of a specific disposal option. Also, the level of detail of FEPs can be dependent of the FEP screening. A guiding principle is to start at an appropriately broad level of detail and then, if during the screening process a FEP is found to be partially included and partially excluded, the FEP may be split into two or more sub-FEPs so that a screening decision on each of the sub-FEPs is non-ambiguous. Experience with the FEPs process permits the identification of FEPs at the appropriate level of detail to minimize such adjustments to the FEP list.

Finally, as noted in Section 1.1, FEP analysis is an iterative process that evolves as new information (e.g., experimental data, model results, socio-political drivers) becomes available. While the preliminary list of 208 FEPs is considered comprehensive, examinations and audits of the UFD FEP list will always be ongoing (e.g., further comparisons with existing and/or newly created FEP lists from other programs).

Similarly, as the FEP list is reviewed and/or utilized by subject matter experts, possible enhancements or modifications may be suggested.

NOTE: THCMBR = thermal-hydrological-chemical-mechanical-biological-radiological

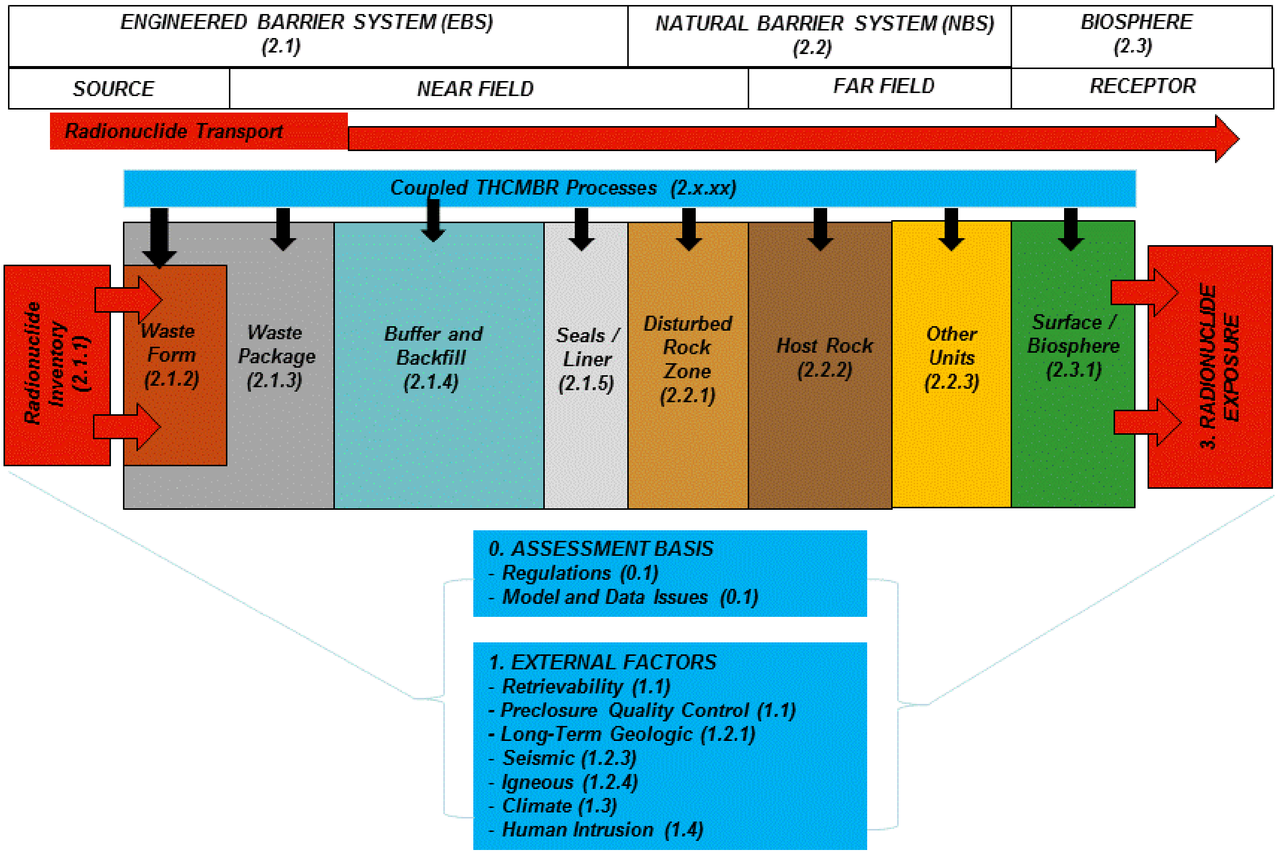


Figure 4-1. Classification and Numbering Hierarchy for UFD Disposal System FEPs

4.2.2.3 Preliminary FEP Screening for the Generic Safety Case

FEP screening requires site characterization and design data, site-specific information about external factors, and regulatory requirements. In this Generic Safety Case, preliminary FEP screening has been performed for each of the four disposal options, based on a number of assumptions about the components and characteristics of the various disposal systems. These preliminary FEP screening decisions are documented in the following reports:

- Generic deep borehole disposal: Brady et al. (2009)
- Generic clay repository: Hansen et al. (2010)
- Generic granite repository: Mariner et al. (2011)
- Generic salt repository: Sevougian et al. (2012).

Preliminary screening decisions were based on differing sets of assumptions for each of the disposal options, and in some cases produced included FEPs that could not be represented with the simplified safety assessment modeling capabilities described in Section 4.3. They were not intended to represent formal screening results, only to provide initial guidance to preliminary research and development and safety assessment model needs. As a result, formal FEP screening to support scenario development for this Generic Safety case was not performed. Instead, the preliminary scenario development described in

Section 4.2.3 was based on simplifying assumptions about the components and characteristics of the four generic disposal systems. These assumptions are documented in Section 4.2.3.3. As new information becomes available from research and development, the FEP screening will be updated and formalized.

4.2.3 Scenario Development

Scenario development consists of the following steps:

1. **Scenario Construction—**Form (construct) scenarios from the retained (included) FEPs, as appropriate
2. **Scenario Screening—**Screen the scenarios using the same criteria applied to FEPs to identify any scenarios that can be excluded from the safety assessment model

A summary of scenario construction and screening approaches and considerations is presented in Section 4.2.3.1. The application of scenario construction to each of the four generic safety case disposal options is described in Section 4.2.3.2. Scenario screening for each of the four safety case disposal options is described in Section 4.2.3.3. Implementation of the screened in scenarios in the safety assessment models is described in Section 4.3.

4.2.3.1 Overview

Overviews of scenario development approaches are presented in Organisation for Economic Co-operation and Development (1992, Sections 5 and 6) and BSC (2005b, Section 2). Summary information is provided in Section 4.2.3.1.1 for scenario construction and Section 4.2.3.1.2 for scenario screening.

4.2.3.1.1 Scenario Construction

Scenario construction involves developing a set of potential scenarios – combinations of included (or retained) FEPs that represent possible future state of the system. Scenario construction (sometimes referred to as scenario formation), therefore, forms a link between the FEPs and the safety assessment models. The goal is to construct a set of scenarios that (1) represents all of the included FEPs, and (2) covers the spectrum of possible future states of the disposal system. Formal FEP screening, together with safety assessment model implementation can be used to support the demonstration of completeness of the safety assessment by providing a structure to ensure that, given a comprehensive list of potentially relevant FEPs, all important FEPs are captured in one or more scenarios.

A common approach to scenario development is to create an undisturbed scenario to represent the expected, or nominal, behavior of the system and one or more disturbed scenarios to represent disruptive events such as human intrusion, seismic events, or igneous events. The undisturbed scenario is often, but not necessarily, considered to be the most likely scenario for the given safety assessment, and may be referred to as the reference or nominal scenario. Disturbed scenarios can be triggered by specific events and may act over different periods of time. Scenarios can also be triggered by a failure of an engineered component (e.g., the failure of a waste package) in the absence of a disruptive event.

In developing scenarios a few principles are followed:

- All included (retained) FEPs must be accounted for in at least one scenario. However, FEPs may appear in more than one scenario (e.g., “nominal” FEPs may also occur as part of disturbed scenarios).
- Scenarios are independent of each other but are not mutually exclusive. In other words, the occurrence or non-occurrence of one scenario has no effect on the probability of occurrence of the other scenarios and the occurrence of one event does not preclude the occurrence of the other events.
- Scenarios must not overlap or consequences may be double counted.
- The process must be documented in a traceable, transparent, and verifiable fashion.

Before scenarios are constructed, it is useful to identify the included FEPs as either expected (those that have a probability of occurrence very close to 1) or disturbed (those that have a probability of occurrence of much less than 1, but greater than any regulatory threshold). The nominal scenario is constructed from the expected FEPs and various disturbed scenarios are constructed from the expected FEPs and combinations of disturbed FEPs. This is not to say that the nominal scenario has a probability of occurrence of 1; most of the expected FEPs are typically part of both the nominal and disturbed scenarios and collectively the probabilities all scenarios sum to 1. Typically the disturbed scenarios have small probabilities of occurrence and the nominal scenario has a probability of occurrence close to 1.

The number of scenarios depends on the resolution at which the FEPs have been defined: coarsely-defined FEPs result in fewer, broad scenarios, whereas narrowly-defined FEPs result in a larger number of focused scenarios. There is no uniquely correct level of detail at which to define scenarios. Decisions regarding the appropriate level of resolution for the analysis are made based on consideration of the importance of the scenario to overall system performance and the resolution desired in the results. As a general rule, scenarios should be aggregated at the coarsest level that provides adequate detail for the purposes of the safety assessment.

Specific scenario development techniques for mapping FEPs to scenarios are summarized in Organisation for Economic Co-operation and Development (1992, Sections 5 and 6) and BSC (2005b, Section 2). Recent international efforts (e.g. Organisation for Economic Co-operation and Development 2012) describe a top-down approach that links FEPs to safety functions. From Organisation for Economic Co-operation and Development (2012, Section 17.7.3):

> *Combining FEPs to scenarios is often called a bottom-up approach whereas the derivation of scenarios by degrading the fulfillment of safety functions is often described as a top-down approach. In practice, both approaches are often used simultaneously in a complementary way; as a result regulations usually do not favour one approach against the other.*

The top-down approach offers additional benefits. The Organisation for Economic Co-operation and Development (2012, Section 5.3) states:

> *Safety functions are useful to describe the initial state and evolution of a system in relation to the safety concept. Scenario sets derived from studying (scientific and technologic) uncertainties potentially affecting the safety functions (e.g. barrier performance) are perhaps not necessarily "complete", but better targeted to, and comprehensive with regard to, safety-relevant issues. However, for providing a sufficient scientific basis concerning the phenomenological knowledge needed to establish scenarios with confidence it will also be necessary to take advantage from systematic and comprehensive databases of the underlying THMC features and processes. Finally, it should be noted that there is also a tendency to formally link the two ways in hybrid approaches, sometimes using formal tools linking FEPs to safety functions.*

4.2.3.1.2 Scenario Screening

Scenario screening approaches are considerations are similar to the FEP screening described in Section 4.2.2.1.2. The set of constructed scenarios can be evaluated against exclusion criteria (e.g., low probability, low consequence, incompatibility with regulatory guidance) to see if any scenarios may be excluded from consideration in the safety assessment.

If scenarios are to be screened out on the basis of low probability, the probability must be taken at an appropriate level. Scenarios should not be defined too narrowly such that their probability of occurrence falls below the regulatory threshold. This can arise for disturbed scenarios where the probability of occurrence is driven by multiple events and the overall probability of occurrence is the product of the individual probabilities of each event occurring.

The scenario screening step is where the probabilities of occurrence and associated scenario uncertainties should be confirmed. For example, in a probabilistic calculation, the expected annual dose may be calculated separately for each scenario and then combined with appropriate probability weighting to estimate total expected annual dose.

Scenario uncertainty is introduced because scenarios attempt to identify plausible future states of the disposal system. To develop a complete understanding of how scenarios might impact disposal safety, uncertainties must be identified, accounted for, and propagated throughout the safety assessment calculations. As described in Section 5.2, this uncertainty in the future state of the system (e.g., the timing or magnitude of the disturbance) is part of aleatory, or irreducible, uncertainty. In constructing the scenario there may also be epistemic uncertainty associated with incomplete knowledge of the data and parameter values and/or model uncertainty.

As with the FEP analyses and scenario construction, it is important to adequately document the scenario screening process, including screening decisions, probabilities of occurrence, and uncertainties, and communicate this information in a transparent fashion to stakeholders.

Once the scenarios have been defined and screened, they are incorporated into the safety assessment models, described in Section 4.3.

4.2.3.2 Scenario Construction for Generic Safety Case

Construction and screening of scenarios generally involves site-specific information as well as specification of waste type, waste form, and disposal system design. As a result, formal construction and screening of scenarios cannot be done generically. Instead, in the absence of formal FEP screening results (see Section 4.2.2.3) and site- and design-specific information, potentially relevant scenarios for each of the four disposal options are described in the following subsections. For each disposal option, the expected initial state is described followed by a discussion of the likely temporal evolution of the generic repository system as a function of the geology of the natural system in which it is sited and of the engineered barriers that are assumed to be present. For each disposal option, the evolution begins at an unperturbed initial state of the host rock, followed by rather abrupt or discrete changes in the environment caused by excavation and waste emplacement, followed by more gradual changes induced by the effect of radioactive decay heat on the natural environment and the tendency to return to the "equilibrium" state in which the system first began, but modified by the presence of the engineered barriers.

Because a specific site and design has not been chosen, the descriptions of the evolution of the generic repositories are concentrated more on undisturbed scenarios, and less on the more site-specific disturbed scenarios. Observations about repository evolution or performance in the event of a human intrusion are included where they provide additional insight about the robustness of a particular repository concept.

The scenarios for each generic disposal option, consistent with the descriptions of the initial state and evolution to final state, are considered for scenario screening in Section 4.2.3.3.

4.2.3.2.1 Generic Salt Repository Scenarios

This section describes (1) the expected initial state of a generic salt disposal system, and (2) the likely temporal evolution to a final state under undisturbed (nominal) conditions. The nominal evolution scenario considers perturbations caused by repository excavation and emplacement of heat-generating waste, including mechanical and hydrologic alteration of the engineered components and near field during and after the thermal pulse. Potential effects from disturbed scenarios, such as human intrusion, are also discussed.

The information is based largely on experience at WIPP (summarized by Hansen and Leigh 2011).

4.2.3.2.1.1 Generic Salt Repository Initial State

As shown in Figure 2-4, the continental U.S. has many salt formations, including bedded and domal salt. Bedded formations of salt (sodium chloride) are found in layers interspersed with materials such as anhydrite, shale, dolomite, and other salts such as potassium chloride (Hansen and Leigh 2011). These formations are tabular and can range across enormous land areas. Bedded salt formations are typically between 200 to 600 m thick, but in some locations in the U.S. they can have thicknesses of up to 1,000 m. Salt domes form from salt beds when the buoyancy effect created by the density contrast between the salt and the overlying rock exceeds the strength of the confining material. Under such conditions, the salt has a tendency to move slowly upward toward the surface, deforming plastically into mushroom-shaped diapirs and many other cylindrical and anticlinal shapes. The top of some domal salt can be near surface, while the root may extend to a great depth. Typically the diameter of a salt dome is on the order of 5 km.

The following initial conditions are characteristic of typical salt formations (U.S. Department of Energy 2011b):

- **Mechanical Properties**
  - Salt flows around buried material and encapsulates it. Salt will slowly deform to surround other materials, thus forming a geologic barrier that isolates waste from the environment. Creep or visco-plastic flow of salt has been well characterized for many applications. Salt domes, for example, rise via buoyancy through overlying sediments without inducing fractures in the salt. If salt is confined, even modestly, it can flow without fractures because the cubic crystal structure and imperfections in the lattice called dislocations afford sufficient slip systems.
  - Fractures in salt are self-healing. In terms of disposal, one of the most important attributes of salt as an isolation medium is its ability to heal damaged areas, such as those created by mining and engineered barrier system construction. The healing mechanisms include microfracture closure and bonding of fracture surfaces. Evidence for healing of fractures in salt has been obtained in laboratory experiments and through observations of natural analogues. Fracture healing can readily restore salt's native low permeability.
- **Hydrologic Properties**—Salt is essentially impermeable and of very low porosity and water content. The very existence of a salt formation millions of years after deposition is evidence that water has not flowed through the formation. The established values for permeability of intact salt come from many industry applications, such as the large-scale storage of hydrocarbon product in solution salt caverns. The native permeability of salt is essentially too low to measure using traditional hydrologic and reservoir engineering methods.
- **Thermal Properties**—Salt has a relatively high thermal conductivity. Thermal conductivity of natural rock salt under ambient conditions is approximately two to three times higher than granite. A relatively high thermal conductivity is a positive attribute in a salt repository for nuclear waste because the radioactive decay heat from used fuel or high-level radioactive waste is rapidly dissipated into the surrounding formation.

These conditions are consistent with the assessment basis developed in Section 3.3.2.1.

In addition to the foregoing attractive initial state properties, salt formations are often very thick, which significantly contributes to waste isolation and containment. Furthermore, excavations in salt are likely to be easier to develop, as compared to clay or granite, which contributes to preclosure safety.

After mining and emplacement of the used fuel and high-level radioactive waste in a salt repository, the natural system's mechanical and hydrologic state has been altered. The primary sources of alteration are (1) the mechanical stress induced by excavation, which results in the fracturing of the host rock in the near-field region around the waste, and (2) communication of the host rock pore environment with the atmosphere, through gas phase transport of water vapor. Also, some heating will occur prior to closure, since waste emplacement and repository closure is not an instantaneous event. The heating effects are discussed in Section 4.2.3.2.1.2 regarding evolution of the repository to a final state. Here, the alterations to the mechanical and hydrologic properties prior to closure are described, which represents the initial state at the time of closure.

A mine layout for disposal of used nuclear fuel and high-level radioactive waste in salt can be quite flexible (e.g., Figure 2-12). A recent conceptual salt repository study called the generic salt repository for high-activity waste advanced a new disposal concept based on lessons learned from the WIPP, Asse, and Morsleben (Carter et al. 2011a). The building block of the underground geometric layout is called a panel, consisting of individual rooms containing a series of over 200 alcoves. This configuration allows emplacement of vitrified waste in the alcoves, with the main room functioning as an access corridor. The disposal strategy assumes placement of one canister at the end of each alcove to be covered by crushed salt backfill for radiation shielding of personnel accessing adjacent alcoves. By providing spacing of 40 ft between adjacent canisters, the areal heat loading of the salt is 39 W/m^2. The thermal loading will accelerate closure of the alcoves and rooms due to thermally-enhanced salt creep.

The DRZ comprises the region near an excavation alcove that experiences changes in hydrologic or mechanical properties. The properties that typically define a DRZ in salt include (Hansen and Leigh 2011): (1) dilational deformation ranging from microscopic to readily visible, (2) loss of strength evidenced by rib spall, floor heave, roof degradation and collapse, and (3) increased fluid permeability via newly created fracture porosity. Salt, being a plastic medium at repository depth, exhibits an isotropic state of lithostatic stress. Excavation of underground openings perturbs the static equilibrium of the mined regions sufficiently to cause fracturing in rock proximal to the excavations, due to deviatoric (shear) states of stress that arise from excavation operations. In salt, shear stress can result in both elastic and inelastic deformation of the host rock. Elastic deformation occurs instantaneously (time-independent) in response to changes in stress state; however, inelastic response in various regions of the DRZ is time-dependent, resulting in visco-plastic flow or “creep.” As described in the Section 4.2.3.2.1.2, it is this inelastic stress-strain response that is responsible for the mechanical evolution of the repository near field to its final state of encapsulation of the waste.

The hydrologic state of the repository after mining and waste emplacement will be determined by the effect of excavation on porosity and fluid permeability. Salt porosity and permeability increase as a result of dilation induced by mining. Intergranular fractures align with the maximal principal stress, increase connectivity, and subsequently increase fracture apertures. Because of the sensitivity between salt permeability and dilation, the DRZ is expected to exhibit high permeability immediately adjacent to the excavation and decrease away from the opening in direct relation to the level of induced damage and deviatoric stresses. The microfracturing will also relieve pore pressure. At the free surface of the excavation opening, the pore pressure will be atmospheric (the gas pressure in the tunnels), whereas the pore pressure will reach lithostatic pressure (approximately 15 MPa at the repository horizon in this example) beyond the DRZ. The pressure gradient through the DRZ promotes brine release into the mined opening, creating a “dry-out” zone adjacent to the drift openings. In particular, during the repository operational period, the accessible brine will tend to migrate down the stress gradient and evaporate into the ventilation air. But, as salt creep closes a disposal room, the stress gradient decreases, fractures heal, and crushed salt (if present) reconsolidates. Thus, conditions in a repository would evolve to significantly limit brine flow toward the waste disposal areas, as well as radionuclide movement away from the disposal containers after they are breached.

The host rock of a bedded salt repository is, typically, dominantly composed of halite (NaCl) with significant amounts of anhydrite ($CaSO_4$), polyhalite ($K_2MgCa_2(SO_4)_4{\bullet}2H_2O$), and clay minerals as minor phases. Salt formation brines tend to have high concentrations of sodium, calcium, and chloride. Lesser amounts of sulfate and carbonate are present. Some brines also have high magnesium concentrations. The in-situ pH of brines is slightly acidic (i.e., about 6.0 to 6.5) and is buffered by the mineral components of the salt formation. Near-field geochemistry in a salt repository is controlled by the potential interactions between the salt formation brine and the waste, packaging, and emplacement materials (Hansen and Leigh 2011). To the extent that brine is available to react with these materials, the pH, oxidation/reduction conditions, gas fugacities, and dominant aqueous species present will evolve over time. The pH of brines after interaction with steel waste packages at low temperatures would probably be similar to that characteristic of the brucite chemical buffer used at WIPP (pH of about 9) (Hansen and Leigh 2011).

4.2.3.2.1.2 Generic Salt Repository Evolution

The presence or absence of brine in the near field is an important factor in the overall evolution of a salt repository (Hansen and Leigh 2011). Near-field geochemistry in a salt repository is controlled by the potential interactions between the salt formation brine and the waste, packaging, and emplacement materials. To the extent that brine is available to react with these materials, the pH, oxidation/reduction conditions, gas fugacities, and dominant aqueous species present will evolve over time. This evolution will be influenced by temperature, especially during the thermal period of a used nuclear fuel and high-level radioactive waste repository in salt. Elevated temperatures cause some aqueous salt solubilities and material reaction rates to increase. These trends apply from ambient temperature up to dry out at 107°C (for NaCl brine at 1 atm). A repository for heat-generating, high-activity waste in salt would be expected to have reducing near-field conditions during the thermal period only if reducing materials such as steel are sufficiently reacted with brine (Hansen and Leigh 2011). Reducing near-field conditions would not be expected if corrosion-resistant nickel-chromium alloys were used for the waste packages (or if no waste packages were used) because of their much lower aqueous reaction rate compared to carbon steel. An additional mechanism for producing reducing conditions is the reaction of brine with UO_2 in the used fuel disposed of in the repository. This is less likely because it would represent a breach of a waste package, but the used nuclear fuel would represent a large mass of reducing material if exposed.

In the WIPP safety assessment model, brine is also essential to corrosion of iron and other metals and for sustained microbial activity. Corrosion of waste packages and other engineered materials in the disposal area of a used nuclear fuel and high-level radioactive waste repository could be enhanced when in contact with concentrated brines at elevated temperatures, and gases will be generated as a result of the corrosion under chemically-reducing conditions. Subsequent to waste package corrosion failure, corrosion of the waste form, its canister, and waste package internal structure materials would occur, releasing radionuclides and generating additional corrosion gases. Combined actions of the corrosion gas generation and decreasing confined space in the disposal area by salt creep deformation would probably pressurize the disposal area, which could result in brine flows and potential transport of dissolved radionuclides away from the disposal areas to some distance. However, in the absence of brine, a salt repository is virtually unaffected by these corrosion processes. This would be the case if a dry-out zone forms around the emplacement areas, as described below.

Because maximal stress differences occur immediately upon creating the opening, the maximum extent of the DRZ manifests quickly, and the brine migration (i.e., dewatering) process begins shortly after excavation. During the operational period, brine will flow down the pressure gradient from the higher (near lithostatic) pressure in the far field to the atmospheric pressure in the mined opening, and decay heat will further mobilize near-field moisture by vaporization, leaving behind salt minerals in the pore space. The initial pore water may be swept away completely by ventilation and possibly condensed elsewhere, away from heat-generating waste, or it may be hygroscopically absorbed by the salt in a cooler area where the relative humidity exceeds 75% (Hansen and Leigh 2011). It is possible that fluid inclusions could migrate through intact salt toward the heat source. However, in the early period when thermal gradients are steep, the migrating inclusions will encounter grain boundaries or microfractures, where they will move down the stress gradient toward the repository ventilation.

Thus, after excavation and waste emplacement, vaporization and fluid migration processes will create a dry-out zone around the waste disposal area, but the duration and extent of the dry-out zone will depend on the waste heat output characteristics, repository thermal loading, and thermal characteristics of the salt.

In particular, as the temperature decreases following the peak heating period, brine will no longer vaporize as it flows toward the waste disposal area, driven by the higher (i.e., near lithostatic) pore pressure in the far field. Thus, although this fluid pressure gradient after closure (i.e., after removal of ventilation) might cause re-saturation of the near-field salt (both backfill salt and DRZ salt), it is likely that the DRZ of a heat-generating waste repository in salt could heal completely, i.e., return to its pre-emplacement state, via creep deformation (facilitated by the thermal perturbation), within a 100-year period (Hansen and Leigh 2011). Temperature effects on salt deformation are dramatic, as shown by laboratory tests on natural salt specimens. Elevated temperature in a salt repository will enhance creep deformation upon placement of the heat-generating waste in the rooms. As salt creep closes a disposal room, the stress gradient decreases, fractures heal, and crushed salt (if present) reconsolidates. Rapid healing of the DRZ would cause the fluid permeability to return to its original state of near zero permeability, thereby preventing brine from re-saturating the salt around the waste package, resulting in a permanent dry-out zone "encapsulating" the package. This dry "halo" would severely reduce the potential effects of changes in brine chemistry induced by the thermal pulse. For example, it would limit corrosion of the waste packages, as well as transport of radionuclides away from the near field (if corrosion does somehow breach the waste packages). Also, without brine in the near field, production of gas through radiolysis would be limited, which would eliminate another possible transport mechanism, i.e., fluid phase pressure in the near field (increased by gas production) would not exceed ambient pore pressures in the far field.

As conjectured by Hansen and Leigh (2011), if models (and/or experiments) indicate that heated salt does not completely encapsulate the waste to create a dry environment, the possibility would remain that brine is accumulated in the near field during the thermal period. This would necessitate that the design basis of a high-activity waste repository in salt include waste packages engineered to survive the thermal period. Brines from the surrounding bedded or domal salt formation or brines introduced by human intrusion would then interact with the waste packages and not with the waste form itself, reducing the chance of radionuclide mobilization during the time of greatest thermal-mechanical property changes (the thermal period). If steel waste packages are emplaced, the oxygen from the excavation and emplacement phase can be consumed by interaction with the brine, and the closed repository could become anoxic. In the absence of oxygen, water could react with steel or other metallic components of the waste packages and produce hydrogen (H_2) gas via anoxic corrosion. Eventually, either oxic or anoxic corrosion of the waste packages could proceed to waste package failure, allowing brines to interact with the waste form. If waste form(s), canisters, and/or backfill materials interact with brines, substantial changes could occur. For example, alteration of oxide wastes or backfill materials, degradation of glass and/or ceramic waste, and corrosion of waste metals or metal canister materials might alter the oxidation state and pH of the near field. These processes may also change the nature of the solubility controlling phases for radionuclide solubilities. The solubility-controlled dissolved concentrations of radionuclides under hypothesized salt repository conditions define the source term for releases (e.g., any sort of direct brine release or advective transport to the biosphere).

For the undisturbed scenario, the foregoing alteration processes may not be important for the waste isolation capability of a salt repository because there is no credible mechanism for movement of radionuclides from a properly sited and sealed salt repository to the biosphere except by diffusion, which is extremely slow. Thus, the only potentially harmful releases are from human intrusion scenarios where flooding of the repository and/or pressure increases lead to releases to the surface through boreholes (Hansen and Leigh 2011; National Academy of Sciences 1996). The robustness of the seal system must be demonstrated in order to build confidence in the repository containment capability during the evolution of the undisturbed scenario.

The human intrusion scenario for WIPP consists of three sub-scenarios that are site specific and depend on the presence of a pressured brine reservoir underneath the repository and whether an exploratory well intersects only the repository or both the repository and the brine reservoir. The probabilities of the sub-scenarios are a function of the local exploratory drilling rates and the size of the brine reservoir, if it exists. In the WIPP safety assessment, cumulative releases from the repository consist almost entirely of direct releases to the surface resulting from unintentional human intrusion by drilling. Direct releases comprise solids removed by the drill (cuttings), material eroded from the borehole walls by the drilling fluid (cavings), contaminated brine (direct brine releases), and pressure-driven releases of solids (spalling). Unless repository pressure exceeds hydrostatic, direct brine releases are zero; spalling releases are also zero unless repository pressure approaches lithostatic. Consequently, the volume of brine in the repository significantly affects releases from the repository through its effect on repository pressure.

As in the nominal scenario, it is expected that the releases from a human-initiated borehole intrusion scenario in a generic bedded salt repository for used nuclear fuel and high-level radioactive waste would be lower because there is less moisture available and lower permeability around the emplaced waste.

4.2.3.2.2 Generic Clay Repository Scenarios

This section describes (1) the expected initial state of a generic clay disposal system, and (2) the likely temporal evolution to a final state under undisturbed (nominal) conditions. The nominal evolution scenario considers perturbations caused by repository excavation and emplacement of heat-generating waste, including mechanical and hydrologic alteration of the engineered components and near field during and after the thermal pulse. Potential effects from disturbed scenarios, such as human intrusion, are also discussed.

The information is based largely on experience from other national repository programs (e.g., France's clay concept (Andra 2005b)).

4.2.3.2.2.1 Generic Clay Repository Initial State

The following initial conditions are characteristic of typical clay formations (Hansen et al. 2010):

- **Hydrologic Properties**—Low hydraulic conductivity, typically 10^{-12} m/s or less
- **Mechanical Properties**—Based on the clay properties assumed by Hansen et al. (2010) for a generic U.S. repository, fractures formed by excavation and heating will close and seal during repository operation and during the first few hundred years after closure. Fluid movement through repository access tunnel and shaft seals will likewise be limited by the low permeability of candidate seal materials.
- **Chemical Environment**—It is expected that a repository excavated in saturated clay formations would have a reducing environment.
- **Physical Properties**—High specific surface area

These conditions are consistent with the assessment basis developed in Section 3.3.2.2.

The foregoing beneficial properties of clay formations have been investigated over the last several decades through extensive research being conducted in a number of countries. The most advanced programs considering clay repository concepts for disposal of spent fuel and high-level radioactive waste are in Belgium (plastic clay), France (argillite), and Switzerland (claystone). The U.S. has many possible clay/shale/argillite basins with similar attributes for permanent disposal, providing a wide array of siting options, as indicated in Figure 2-5.

Potentially suitable mudstone, clay, shale, and argillite formations share many characteristics favorable to repository development and waste isolation (Hansen et al. 2010). For example, these types of sedimentary media have persisted tens of millions to hundreds of millions of years in almost all geologic provinces in the U.S., and have mostly remained in the same state that was attained by lithification (Gonzales and Johnson 1984).

A comprehensive study by the "Clay Club" (Organisation for Economic Co-operation and Development 1996) described the basic physical and chemical processes that combine to control the flow of water, gas, and solute through clay media. Much is known about these processes, but without a specific site, a range of properties must be considered for a generic clay repository in the U.S. The generic clay repository concept considered by Hansen et al. (2010) drew upon the three aforementioned clay concepts being considered internationally, and assumed that these three are reasonably representative of potential host rocks in the U.S. These three sites are:

- The Opalinus claystone at Mont Terri, Switzerland
- The Callovo-Oxfordian argillite/mudstone near Bure, in eastern France
- The Boom clay at Mol, Belgium

Table 4-1 summarizes some of the most pertinent mechanical and hydrologic properties for these three well-characterized clay formations. The most important of these formation characteristics with respect to long-term performance are low hydraulic conductivity and high sorption. For example, as stated by Andra (2005b): "Given the low permeability of the Callovo-Oxfordian formation (5×10^{-13} to 5×10^{-14} m/s on average), these water flows are very low (a few hundredths of a milliliter per year per meter squared) as is their velocity (approximately a few centimeters per 100,000 years). In this context, the transport of solutes in the Callovo-Oxfordian layer takes place mostly by diffusion." Regarding mechanical properties, if the formation exhibits a brittle style of deformation in laboratory and field-scale tests, then advective fluid flow may occur in fractures induced by excavation or heating. If the host material is plastic, fracture flow is less important because fractures tend to seal or heal, and diffusion becomes the dominant transport process. To a certain extent, engineering can mitigate operational difficulties associated with weaker rock types, so strength properties are not critical (Hansen et al. 2010). The same repository concept (layout, packaging, etc.) can be applied to both types of lithology.

Table 4-1. Typical Properties of Well-Characterized Clay Formations

Formation	Approximate Geologic Age (Ma)	Typical Thickness (m)	Clay Content (wt. %)	Classification	Mineralogy	Carbonate Content (wt. %)	Hydraulic Conductivity (m/s)	Compressive Strength (MPa)	Organic Content (wt. %)	In-situ Water Content (vol. %)
Opalinus Clay	180	160	50–65	Claystone	Kaolinite, illite, illite/ smectite	10–50	Est. 5×10^{-13} – 6×10^{-14}	12	0.5	4–6
Callovo-Oxfordian Argillite	155	130	45	Mudstone	Illite/ smectite	20–30	Est. 3×10^{-14}	25	< 3	5–8
Boom Clay	30	100	55	Bedded mud	Smectite /illite	1–5	Est. 6×10^{-12}	2	1–5	22–27

Source: Hansen et al. 2010; Andra 2005b; Hansen and Vogt 1987; Nagra 2002; Organisation for Economic Co-operation and Development 2003b; Neuzil 2000; Volckaert et al. 2005.

Table 4-1 indicates typical pre-excavation host rock properties. However, excavation and construction effects, ventilation, and the thermal pulse will lead to changes in these rock properties prior to repository closure, including clay dehydration and deformation. Based on prior research and modeling, these effects will be confined to an DRZ within a few meters of the repository and within a few centuries after waste emplacement, overburden pressures will seal fractures, resaturate the dehydrated zones, and provide a repository setting that strongly limits radionuclide movement to diffusive transport (Hansen et al. 2010), as described below in Section 4.2.3.2.2.2. As stated by Andra (2005b): "The hydraulic disturbance caused by the repository remains limited to the repository itself and the Callovo-Oxfordian formation on account of its low permeability. It disappears after approximately 100,000 years and a new state of hydraulic equilibrium is then established in the repository and the Callovo-Oxfordian layer."

To characterize these effects of construction on the initial host rock state, some engineered barrier system design information is necessary. To select design information and clay properties for a generic U.S. repository, Hansen et al. (2010) examined advanced repository programs in other countries that are focused on clay media. Based on disposal concepts in Europe, disposal operations for a generic repository would be horizontal. For example, the French program opted for horizontal placement because vertical handling would require additional overhead space, and horizontal waste handling allows more compact design. This concept of operations, along with the waste form characteristics (which depend on the duration of surface storage prior to emplacement in the repository), will determine the repository areal footprint. For example, the French design for disposal of vitrified high-level radioactive waste ("C waste") maintains a distance between two adjacent horizontal disposal boreholes sufficient to meet a thermal management requirement that limits the temperature to 90°C in the geologic medium (Andra 2005b). This basic design layout results in a borehole spacing distance of at least ten times the diameter of the emplacement boreholes. Also, by orienting the boreholes parallel to the maximum principal stress direction, fracturing and microfissuring in the DRZ is minimized.

As noted by Hansen et al. (2010), comparison studies of programs in other countries performed to date (U.S. Nuclear Waste Technical Review Board 2009) have concluded that the long-term performance of an engineered barrier system is unimportant to the safety case for clay repositories. However, it can be important for retrievability and reversibility during the operational phase. For example, the French engineered barrier system design consists of vitrified waste placed in stainless steel containers within a steel overpack. The steel overpack was chosen for high-level radioactive waste disposal to "prevent water reaching the glass for a period of several thousand years," in order to "avoid the risk of piercing the primary stainless steel container by corrosion through water contact…and a dissemination of radioactive nuclides which would make package retrievability operations trickier (reversible management)," and to "prevent early alteration of the glass, accelerated by the temperature, which would be accompanied by radionuclides release (safety)." (Andra 2005b).

Other clay repositories described by Hansen et al. (2010) also include steel containers as part of the engineered barrier system. For example, the Belgian "supercontainer" concept envisions an engineered barrier consisting of stainless steel canisters holding high-level radioactive waste inside a carbon steel overpack surrounded by thick concrete. The Swiss engineered barrier concept envisions cast iron canisters for spent nuclear fuel and high-level radioactive waste, where high-level radioactive waste is contained in a stainless steel flask inside the canister. The iron canisters are to be surrounded by bentonite clay. Furthermore, both the Belgian and Swiss concepts include a backfill around the waste packages, in order to enhance the waste isolation capability of the engineered barrier system and, in the case of the Belgian repository, to provide structural support for the emplacement tunnels. The Swiss concept will install waste canisters in emplacement tunnels, which will then be backfilled. The waste canisters will be placed on highly compacted bentonite clay blocks, co-axially with the tunnel axis, with a distance of 3 m between canisters. No steel liner is currently designed for the horizontal disposal concept because of favorable rock mechanical properties. The clay buffer material used to backfill the tunnels will act as an extra measure of compartmentalization and isolation of the waste canisters. Retrievability is believed possible, but is not the highest priority design requirement. On the other hand, in Belgium, the repository concept is adapted to a geologic formation with less mechanical strength, so the disposal galleries in the Boom clay will be structurally supported by a liner built from circumferential concrete wedge blocks.

Seal systems are also an important part of the engineered barrier, and the fundamental design principle for seal systems in a clay repository is to ensure that radionuclide transport is controlled by diffusive rather than advective processes. The shaft seal system would limit entry of formation water into the repository and restrict the release of fluids that might carry contaminants. Seals are designed to limit fluid transport through the opening itself, along the interface between the seal material and the host rock, and within the disturbed rock surrounding the opening. Given the timescale for natural resaturation of swelling clay seal materials, corresponding activities must be developed to provide evidence that seals will function as designed, without having to monitor the seals in their long-term configurations.

The DRZ for any excavation is created by changes to the preexisting stress state and is a function of the material properties in relation to the stress conditions. Fractures, in particular, are perhaps the most important excavation effect, because they can have an appreciable effect on the isolation capability of the host rock formation due to the increase in fluid permeability. In clay media, rock fractures are not only induced directly by mechanical stresses from the tunnel openings but also by near-field desaturation and dessication that might lead to local fracturing and material weakening. As described by Andra (2005b), in the French concept, fractures are of two types around the drift openings: (1) *a deconfinement-induced fractured zone* in the immediate vicinity of the excavation, produced if the maximum mechanical strength of the rock is exceeded, and characterized by the appearance of more or less connected fractures parallel to the drift axis; and (2) *a microfissured zone*, formed when the fissuring threshold is exceeded, either immediately at the excavation wall (if the fractured zone has not yet developed), or behind the fractured zone. Based on in-situ testing, the permeability in both zones is still very low, with the fractured zone having a hydraulic conductivity less than 10^{-8} m/s and the microfissured zone having a conductivity less than 10^{-12} m/s. Regarding the potential effect of desaturation on material strength, in the Callovo-Oxfordian argillite being considered in France, testing has shown that its effect is quite limited.

Safety assessments for clay repositories, such as that described in Section 4.3.2.2, must consider these excavation perturbations to the host rock, and especially changes to fluid permeability, and whether postclosure evolution of the host rock will tend to return the disturbed rock properties to their initial state.

4.2.3.2.2.2 Generic Clay Repository Evolution

Clays naturally have a significant water content (Table 4-1), which will influence the evolution of processes that initiate in response to repository excavation and waste-form decay heating. These responses include desiccation, alteration of pore-fluid chemistry and mineralization, and mechanical weakening and/or creep. These specific features and processes need to be considered in concert with the geometry, thickness, depth, and stability of the repository host rock, to assess long-term containment of radionuclides. Safety assessments for clay repositories must consider whether increased permeability in the DRZ, through fracturing and fissuring, could support a pathway for radionuclide transport either to the radial limit of the zone, or along the DRZ parallel to the excavated openings.

Excavation in many geologic media causes mechanical damage to the rock around the opening, which forms and then stabilizes during excavation (Hansen et al. 2010). The initial extent of the DRZ around openings in clay formations is determined by rock strength and deformation properties, initial liquid saturation, in-situ stress conditions, the opening size, and the resulting pore pressure changes. These processes will produce micro- and macro-scale fractures that increase permeability (Nagra 2002). Partial desaturation is likely to occur by evaporation during ventilation for a few years during the operational phase of a repository, and cause shrinkage and stiffening of the clay, and resistance to creep (Andra 2005b). After the emplacement openings are sealed, resaturation will occur gradually (requiring perhaps 100,000 to 200,000 years), accompanied by swelling and creep of the buffer and disturbed host rock. Thus, the increased permeability will eventually be reversed, at least in part, by "sealing" processes, which include swelling, disintegration, creep, and consolidation (Bernier et al. 2007). Hence, although fractures may be formed and/or opened by stress changes and deformation in the DRZ, if no permanent alteration occurs they can subsequently close after resaturation, with low permeability re-established through self-sealing of the DRZ, assuming temperatures are kept below 100°C as described in Section 2.2.3.3.2.

The initial rock hydraulic conductivity (Table 4-1) of the DRZ near the openings for a repository in clay would be increased by orders of magnitude compared to the initial undisturbed conductivity, to as much as 10^{-8} m/s, followed by reduction due to sealing over the next few years. Sealing of fractures and other voids that form during excavation will cause the final hydraulic conductivity of the DRZ to be less than

10^{-12} m/s after a few years (Hansen et al. 2010), especially if the opening is backfilled with a swelling material such as bentonite (considering the Opalinus clay to be representative; Blümling et al. 2007). This is effectively the same as the initial, undisturbed permeability of clay, which is so small that liquid-phase advective transport is negligible, which implies that radionuclide migration in the long term will be effectively limited to very slow diffusive transport through the (mostly undisturbed) clay. According to Andra (2005b), “creep, accompanied by the resaturation of the argillites, gives rise to the closure of any fracturing of the rock and compresses the micro-fracturing of the damaged zone. This gradual healing of the rock tends to re-establish a degree of permeability close to the sound rock one.” While this healing process is expected for more plastic clay formations, it is not necessarily true for indurated shales.

Heating of the host rock will begin as soon as heat-generating waste is emplaced. Thermal response will be dominated by thermal expansion, principally of the pore water, which has higher thermal expansivity than the solid framework. The resulting elevated pore pressure, combined with stress redistribution near excavated openings, reduces strength and increases deformation in clay media. However, the potential contribution from thermal effects to the nature and extent of the DRZ is likely to be negligible if maximum temperature is limited, e.g., less than 100°C (Hansen et al. 2010). As indicated by Andra (2005b): “Given the limited duration of the thermal processes and the maximum temperatures reached in the various zones of the repository, the argillite layers initial properties and the repository components characteristics are not, or only to a small extent, affected. In particular, the mineralogical transformations in the Callovo-Oxfordian formation are small.”

For the nominal, undisturbed scenario, the largest driving force for fluid flow and radionuclide migration away from a clay repository may be thermally induced pore pressure transients. However, the engineered barriers will be designed to function through and well beyond the duration of the thermal pulse to prevent radionuclide migration during this period, in the case of a prematurely breached canister. Furthermore, the thermal peak will occur within a few hundred years, after which water must return to the repository to mobilize radionuclides, but can do so only after cooling, and after rehydration.

After the decay-heat thermal pulse, if drift closures and shaft seal systems perform as expected, radionuclide transport to and within the far-field host rock would be limited by low permeability, so that diffusion is the dominant transport mechanism for 1,000,000 years or longer. Both liquid and gaseous advection are limited by intrinsic permeability on the order of 10^{-19} m^2 or less. For example, the presence of overpressured fluid in the Opalinus clay (estimated to have head as much as 100 m greater than a water column to the ground surface (Nagra 2002)) gives strong evidence of low permeability. For this excess head to persist over geologic time signifies that advective transport is not significant in the repository timeframe.

Based on the above consideration and an evaluation of appropriate FEPs for a clay repository, Hansen et al. (2010) developed three scenarios whereby radionuclides could be transported to a hypothetical aquifer, from which they would be pumped to the biosphere: (1) short-term, advective transport through the repository openings or the DRZ, and up the shafts; (2) long-term, diffusive transport through the host clay upward from the emplacement boreholes; and (3) a stylized human intrusion scenario.

It was decided that the first scenario is not of much consequence because of its short-term nature, the likely effectiveness of engineered seals, and the lack of a strong hydraulic pressure gradient to drive water through the repository and up the shafts.

For the second scenario, thermal, hydrologic, and geochemical calculations suggest that radionuclides in a clay repository will not migrate far from the disposal horizon (Hansen et al. 2010). The reducing environment typical for clays would limit radionuclide solubility, thus limiting mobility. Also, the minerals present in clay formations readily sorb many radionuclides, further attenuating releases. Much of the inventory will decay before transport to the biosphere can occur. The calculated dose was found to be significantly less than any plausible regulatory limits, and this was based on assumptions such as instantaneously degraded waste packages and waste forms, unlimited availability of moisture for waste form degradation and transport, and no sorption on degraded waste package materials.

The third scenario is highly dependent on specific siting and design considerations. A human intrusion scenario in clay would likely involve advective transport up one or more sealed or unsealed boreholes drilled in the future after repository closure (e.g., a borehole for hydrocarbon exploration drilled through the repository and later abandoned). A vertical hydrologic gradient would transport radionuclides to a shallow aquifer from which they would be pumped to the biosphere. Probabilities of occurrence of human intrusion are site- and regulation-specific.

4.2.3.2.3 Generic Granite Repository Scenarios

This section describes (1) the expected initial state of a generic granite disposal system, and (2) the likely temporal evolution to a final state under undisturbed (nominal) conditions. The nominal evolution scenario considers perturbations caused by repository excavation and emplacement of heat-generating waste, including mechanical and hydrologic alteration of the engineered components and near field during and after the thermal pulse. Potential effects from disturbed scenarios, such as glaciation, are also discussed.

The information is based largely on experience from other national repository programs (e.g., Sweden's granite concept (SKB 2006b) and Finland's granite concept (Posiva 2010b)).

4.2.3.2.3.1 Generic Granite Repository Initial State

Granite is common across the U.S. at depths suitable for a mined repository for nuclear waste. A repository in granite that incorporates the Swedish KBS-3 repository concept (SKB 2006b) is anticipated to satisfy safety criteria. However, unlike the safety cases for disposal in salt, clay, or deep boreholes, the safety case for a mined granite repository depends strongly on waste package performance (Mariner et al. 2011). In crystalline rock, waste packages are preserved by the high mechanical stability of the excavations, the diffusive barrier of the buffer, and favorable chemical conditions. The buffer is preserved by low groundwater fluxes, favorable chemical conditions, backfill, and the rigid confines of the host rock. Because of this ability of a granite repository to preserve waste canisters, canister failure in less than one million years would have a low probability, affecting a small fraction of the waste packages. For those that fail, slow waste form degradation, low solubility, and sorption to buffer materials would substantially delay release of much of the radionuclide inventory from the repository and allow for significant radioactive decay. Beyond the repository, the low permeability of the granite host rock would strongly inhibit radionuclide transport to the biosphere. An added advantage of a mined granite repository is that waste packages would be fairly easy to retrieve, should retrievability be an important objective. Granitic formations are extensive in the U.S.; surface outcrops of granite are shown in Figure 2-6 and depth to basement crystalline rock in the U.S. is shown in Figure 2-7.

Granitic repository programs in Europe (e.g., Sweden and Finland), North America (Canada), and Asia (Japan), which have matured over the past three decades and have generated large sets of data from laboratory experiments and in-situ tests performed in underground research laboratories, provide a valuable resource for investigating a potential U.S. repository in granitic or crystalline host rock. A full-scale emplacement demonstration in an underground research laboratory at the Climax Stock on the Nevada National Security Site also demonstrated the feasibility and safety of spent fuel storage and retrieval from an underground facility in granite (Patrick 1986). Two of these programs have either entered the license application phase (Forsmark, Sweden) or are very close to doing so (Olkiluoto, Finland). For the generic repository described here, design and operational concepts from these other programs are used as the technical basis. This use of existing work permits a credible analysis that applies to granitic formations in the U.S. that have the potential to host a repository for used nuclear fuel and high-level radioactive waste.

The following initial conditions are characteristic of typical granite formations (Rechard et al. 2011; Mariner et al. 2011):

- **Mechanical Properties**—High mechanical strength and minimal rock pressure
- **Thermal Properties**—High heat resistance and moderate thermal conductivity. Granite is composed largely of silicate minerals, which crystallize at high temperatures. Therefore, it is unlikely that the silicates would be significantly affected by the heat generated from radioactive waste.
- **Hydrologic Properties**—Low permeability and low water content. The rock matrix permeability of granitic rocks is very low (comparable to clay media) and most of the hydraulic conductivity is associated with interconnectivity of fracture networks, which tends to decrease with depth. In granite plutons, lighter freshwater is found near the surface and heavier saline water is found at depth.
- **Chemical Properties**—It is expected that a repository excavated in saturated granite formations would have a reducing environment.

These conditions are consistent with the assessment basis developed in Section 3.3.2.3.

Table 4-2 summarizes some of the pertinent hydrologic and mechanical properties for well-characterized granite formations in the Canadian, Swedish, and Finnish programs. These data show that rock strength is high and porosity is generally less than 1%. The wide range of hydraulic conductivity observed is highly correlated with the number, aperture, and connectivity of fractures. Granite bodies are intersected by fracture zones of varying hydraulically conductivity; thus, the higher measurements reflect the effects of fracture zones. The decrease in hydraulic conductivity with depth is consistent with observations of fewer open, interconnected fractures at depth.

Table 4-2. Typical Properties of Well-Characterized Granite Formations

Location	Reference	Lithologic Classification	Porosity (%)	Bulk Hydraulic Conductivity (m/s)	Young's Modulus (GPa)	Poisson's Ratio	Compressive Strength (MPa)
Atikokan, Ontario	Gascoyne et al. (1987)	Granite		10^{-12} – 10^{-5} (<400 m) 10^{-13} – 10^{-10} (>400 m)			
Olkiluoto, Finland	Posiva (2010b)[a]	Pegmatitic granite	0.01–0.2	10^{-7} (near surface) 10^{-11} (at depth)	65	0.29	108
Laxemar, Sweden	SKB (2006a, Table A-42); SKB (2006b)	Granite to granodiorite	0.14	10^{-10} – 10^{-5} (<200 m) 10^{-11} – 10^{-6} (>200 m)			165–210
Forsmark, Sweden	SKB (2006a, Table A-42)	Granite to granodiorite	0.09		76	0.24	

NOTE: [a] Approximate range of flow porosities, which are 1/10th the approximate diffusion porosities (Tables C-3, C-4, and C-5 ; Posiva (2010b)). Hydraulic conductivity values are from Posiva (2010b, p. 249).

Similar to the hydrologic and mechanical properties, the chemical environment in the repository host rock is important for a number of reasons, as alluded to above, related to performance of both the engineered barrier system (e.g., waste package corrosion, radionuclide solubility and mobilization) and the natural barrier system (e.g., sorption capacity). At the depth of a potential repository in granite (approximately 500 m), brackish groundwater typically saturates the fractures and interconnected pores (Mariner et al. 2011), and the redox conditions are reducing, which is beneficial because these conditions prevent rapid oxidation of waste canisters, decrease aqueous solubilities of many redox-sensitive radionuclides, and can limit waste form degradation rates. For example, in Canada, all Eh measurements from groundwater samples from four research areas in the Canadian Shield indicate redox potentials at or below the Fe(II)/Fe(III) boundary (Gascoyne et al. 1987).

As with any mined repository, excavation and construction effects, ventilation, and the thermal pulse can lead to changes in granite host rock properties in the near field prior to repository closure. This affects the initial state of the repository at the time of closure; to characterize this state it is necessary to address repository and engineered barrier design. Because Sweden, Finland, and Canada have advanced concepts for a repository in granite, their experience was used by Mariner et al. (2011) as a basis for a generic repository design in a U.S. host rock granite. The conceptual granite repositories for these countries are at depths of 420 m (Finland) and 500 m (Sweden and Canada) (SKB 2006b; Posiva 2010b; Garisto et al. 2009). The reference layout and arrangement of waste packages is similar among the three countries. Sweden and Finland are studying waste package emplacement in vertical boreholes drilled into the floor in horizontal emplacement tunnels but are also considering emplacement in horizontal boreholes drilled in the tunnel walls. Canada is considering these two options in addition to in-room horizontal emplacement (Gierszewski et al. 2004). Each of these concepts includes a clay buffer surrounding the waste packages, a mixture of clay and crushed rock to be used as backfill, and various grouting and sealing processes. Because the temperature in the repository will be elevated during the first few thousand years, waste package thermal loads and spacing within the repositories at the time of emplacement are designed in most programs to prevent temperatures at the waste package surface from exceeding 100°C. This limit is primarily imposed to prevent damage to the clay buffer due to alteration of the dominant clay mineral from montmorillonite to illite (SKB 2011). Conversion to illite eliminates the swelling function of the buffer clays, the purpose of which is to create a low permeability seal around the waste canister (SKB 2010). To satisfy this thermal requirement in the Forsmark repository design, the emplacement tunnels are spaced about 40 m apart and have vertical disposal boreholes about 6 m apart (SKB 2006b).

Similar to their role in clay repository concepts, engineered barriers have an important role for waste isolation in fractured, crystalline rock (Mariner et al. 2011). Whereas the natural setting may attenuate sorptive or redox-sensitive waste species that are released from the repository, the more mobile species (e.g., ^{129}I) may be transported rapidly by fracture flow. Thus, the KBS-3 disposal concept (SKB 2006b), developed by the Swedish repository program, calls for disposal of spent fuel in long-lasting, corrosion-resistant copper canisters (1.05-m diameter), with a cast-iron insert for strength and shielding. The copper canisters are to be emplaced in large-diameter (1.75 m) deposition boreholes drilled into the floor in access tunnels (5.5-m diameter) approximately 500 m below the surface. The space between the canister and borehole walls is filled with a low-permeability buffer material containing bentonite or other swelling clay emplaced initially in its dry, compacted form. The principal buffer function is to limit the transport of aqueous solutes between the waste package and the geosphere. Molecular diffusion is the dominant transport process. For many radionuclides and other solutes, transport through the buffer is further limited by sorption. The buffer material contains expansive clay as a major constituent and is emplaced in the compacted, dry state. Buffer swelling and transport performance are only slightly or moderately sensitive to temperature and saline solutions (i.e., seawater). Once hydrated the buffer not only limits outward transport but also limits the rates of inward transport of dissolved reactants that control degradation of the canister and its contents.

Section 4.2.3.2.3.2 describes the evolution of these engineered materials, as well as the DRZ region of the host rock, and the far-field region of the host rock (via glaciation) from this initial state to a state that will exist following the decay-heat thermal pulse and hydrologic-mechanical re-equilibration of the DRZ.

4.2.3.2.3.2 Generic Granite Repository Evolution

In a typical granitic or crystalline host rock repository, chemically-reducing conditions, buffer-limited advection of groundwater, and long-term stability of excavations would provide a physical and chemical environment that would likely preserve copper waste canisters for millions of years. This is important for limiting releases of radionuclides to the accessible environment because subsequent to diffusion through the bentonite buffer, radionuclides may be transported relatively quickly through the fracture network of the granite host rock.

This section summarizes the expected evolution of the repository over one million years. The text presented here is taken directly from Mariner et al. (2011), which was based on Gierszewski et al. (2004) and McMurry et al. (2003). Additional information originated from calculations conducted specifically for the safety assessment of a generic repository analyzed by Mariner et al. (2011), as well as from reports issued by the Swedish (SKB) and Finnish (Posiva) programs.

0 to 100 Years—Repository tunnels, rooms, and/or emplacement boreholes would be excavated during a period of approximately 30 years. A DRZ would form within a few tens of centimeters of these excavations but would not form a continuous pathway for water movement (SKB 2006b). Waste packages would be emplaced and surrounded by a clay buffer. Emplacement tunnels or rooms would be backfilled and sealed, but access tunnels would remain open. During this period, temperatures within the repository would peak (less than 100°C by design) and begin to decline (SKB 2006b; Posiva 2006). The initially low hydraulic pressures in the tunnels and clay buffer would induce water migration toward the waste packages, causing the clay buffer to swell as it hydrates. Minor corrosion of the copper canisters would occur due to the presence of oxygen but after backfilling, the oxygen would be limited; so, copper corrosion by direct reaction with oxygen would be limited to a few tens of micrometers (SKB 2006b; Posiva 2006). Heat from the waste packages would warm the compacted clay buffer from the inside out. Any water vapor produced by evaporation near the waste package would condense further away where temperatures would be cooler. Hydration would thereby proceed inward. A small increase in alkalinity could result from groundwater interaction with cementitious engineered materials, such as grout, but the effects would be negligible (SKB 2006b). Microbial activity would be inhibited within the hydrated clay buffer by lack of nutrients, swelling pressure, and the attendant small pore size.

100 to 1,000 Years—At approximately 100 years, the repository would likely be closed and all access tunnels and shafts would be backfilled and sealed. Large thermal, hydraulic, chemical, and mechanical gradients would slowly dissipate as heat, water, and solutes disperse and swelling of the buffer helps to redistribute rock stresses. Temperatures in the repository would decline but remain elevated (SKB 2006b, Section 9.3.4), while air in the repository would be replaced by water. Any oxygen introduced during excavation and emplacement would be consumed by organic and inorganic processes, causing reducing conditions to be established throughout the repository, which would significantly limit corrosion rates. Buffer material surrounding the waste packages and mixed into the backfill would swell as it hydrates, distributing loads across repository components. Loads from the surrounding rock would be transmitted through the backfill and buffer onto the container. These loads would be high enough to compress the copper canister onto the steel inner vessel but too low to deform the inner vessel. They would also be high enough to eliminate microbes and prevent canister sinking (SKB 2006b; Posiva 2006).

1,000 to 10,000 Years—The large thermal and hydraulic gradients of the first one thousand years would continue to dissipate, but more slowly as the geosphere absorbs and disperses the effects of the repository. Temperatures in the repository would slowly decline and level off at near ambient values near the end of this period (SKB 2006b; Posiva 2006), while cementitious seals and plugs would degrade allowing new flow pathways to develop (Posiva 2006). However, the hydrated buffer material surrounding the waste packages would prevent significant advection of water, dissolved solutes, and colloids through the buffer. Corrosion of the copper canister would be limited by lack of oxygen and a very low flux of other reactants, most notably hydrogen sulfide, through the hydrated buffer to the waste package surface (SKB 2006b).

10,000 to 100,000 Years—From 10,000 to 100,000 years, the thermal pulse would be dispersed and temperatures within the repository (and within the waste packages) would nearly equal the temperature of the original host rock (SKB 2006b). By the end of this time interval, the earth could be in the middle of the next glacial period. Depending on the latitude of the repository, permafrost followed by an ice sheet or glaciers could advance over the ground surface above the repository. Although permafrost could develop near the surface as the mean surface temperature decreases, freezing temperatures would not likely reach the repository (SKB 2006b). Additional rock stresses, creep deformation, and fracturing around the repository due to the increased load of a potential glacier or ice sheet would not likely significantly affect waste package integrity or repository performance (SKB 2006b). If the repository were located in the northern U.S. latitudes, groundwater flow rates at the repository level would decrease with the onset of permafrost at the ground surface due to decreasing infiltration. This effect could be counteracted by the presence of glaciers or a possible ice sheet. If the repository were located in the southern U.S., increased rainfall could increase the circulation of deep groundwaters. Corrosion of the copper outer barrier would continue to be limited and would not threaten containment as long as the buffer surrounding the waste package remained intact (SKB 2006b). Some replacement of sodium in the buffer material by calcium from groundwater and by dissolution of gypsum in the buffer would occur, but the extent would be limited and would not significantly affect buffer performance.

100,000 to 1,000,000 Years—For repositories located in northern latitudes, glacial periods could occur on a regular basis during this time interval. Based on multiple lines of evidence, including ice core data from Greenland and Antarctica, a glacial phase has occurred on average every 120,000 years over the past 650,000 years. However, major changes in ocean circulation, atmospheric composition, or the earth's crust could potentially disrupt the glacial cycle. Retreating ice, which could occur multiple times during glacial periods, would enhance deep circulation of groundwater and could potentially cause the erosion of buffer materials surrounding waste packages. Large vertical movements in the host rock could occur as a result of ice loads, slow crustal movements, and associated seismic events. Shear movements along existing fractures intersecting the repository could cause a small number of canister failures. The probability of such failures is expected to be low partly due to active avoidance of significant fractures during waste emplacement (Posiva 2006). Hydraulic erosion of the buffer during periods of deep groundwater circulation could expose a small fraction of the waste packages to advective groundwater flow (SKB estimates this fraction to be 0.004 within the reference evolution of its license application (SKB 2011)). This exposure would enhance corrosion rates by allowing a larger influx of corrosive reactants like hydrogen sulfide. In addition, erosion of the buffer would allow sulfate-reducing bacteria to live near the waste package and generate hydrogen sulfide from an advective influx of sulfate. A small number of copper canisters exposed to advective flow (due to buffer erosion) could fail within one million years. Degradation of the canister insert and the waste form would begin soon thereafter, and radionuclides would be released from these waste packages. These glacial effects are much less likely to impact disposal systems in areas distant to northern zones of ice accumulation.

4.2.3.2.4 Generic Deep Borehole Disposal Scenarios

This section describes (1) the expected initial state of a generic deep borehole disposal system, and (2) the likely temporal evolution to a final state under undisturbed (nominal) conditions. The nominal evolution scenario considers perturbations caused by borehole drilling and emplacement of heat-generating waste, including mechanical and hydrologic alteration of the engineered components during and after the thermal pulse.

The information is based largely on preliminary conceptual work at the Massachusetts Institute of Technology and Sandia National Laboratories (Brady et al. 2009).

4.2.3.2.4.1 Generic Deep Borehole Initial State

The deep borehole disposal concept consists of drilling deep boreholes into crystalline rocks for permanent disposal of used nuclear fuel and high-level radioactive waste. The disposal concept relies on the presence of crystalline basement rock (e.g., granite) at many stable continental locations, and on existing drilling technology to construct boreholes at an acceptable cost.

The following initial conditions are characteristic of typical crystalline rocks (Brady et al. 2009):

- **Rock Type**—Desirable crystalline basement rocks are relatively common at 2- to 5-km depth, as shown in Figure 2-7
- **Mechanical Properties**—Crystalline rocks such as granites are particularly attractive for borehole emplacement because of their large size, relatively homogeneous nature, low permeability and porosity, and high mechanical strength (to resist borehole deformation). In addition, high overburden pressures contribute to sealing of some of the fractures that provide transport pathways.
- **Thermal Properties**—High heat resistance and moderate thermal conductivity
- **Hydrologic Properties**—Deep crystalline rocks typically have low porosities (< 1%), very low permeabilities (10^{-16} – 10^{-20} m^2), and low water content. Minimal flow is thought to occur, primarily through discontinuous fractures. Several lines of evidence indicate that groundwater at depths of several kilometers in continental crystalline basement rocks has long residence times and low velocity. Basement rocks do not typically contain pressurized aquifers or other flow features that would produce significant upward flow gradients under ambient conditions.
- **Chemical Properties**—High salinity fluids and reducing environment.

These conditions are consistent with the assessment basis developed in Section 3.3.2.4. Table 4-3 summarizes some typical characteristics of deep borehole locations.

Table 4-3. Typical Deep Borehole Characteristics

Borehole	Maximum Depth of Water Circulation (m)	Minimum Depth to High Salinity Water (m)	Permeability below 1000 m (m^2)
USA-10	900	1800	10^{-18}
FRG-2	500	3500	Not Reported
SWT-1	1050	1326	10^{-16} – 10^{-20}
URS-1	800	1200	10^{-19}
SWE-1	1200	>6000	10^{-16} – 10^{-17}

Source: Brady et al. 2009, from Juhlin and Sandstedt 1989.

Numerous factors suggest that deep borehole disposal of high-activity waste is inherently safe but, as with mined waste repositories, the excavation (in this case, drilling) activities introduce mechanical, chemical, and hydrologic perturbations to the native host rock, which must be accounted for when establishing the initial repository state prior to permanent closure. In addition, the decay heat from the emplaced waste introduces a thermal perturbation. To understand these perturbations, a generic emplacement design must be assumed. As suggested by Brady et al. (2009), waste is envisioned to be emplaced as fuel assemblies stacked inside drill casing that is lowered and emplaced, using off-the-shelf oilfield and geothermal drilling techniques, into the lower 1-2 km portion of a 3-5 km deep, vertical, ~ 45 cm diameter borehole. Waste emplacement is followed by sealing the 1 km borehole length above the waste and plugging and backfilling the upper 2 km of the borehole.

In a more specific "reference" design, Arnold et al. (2011) envision waste canisters emplaced in strings of 40 canisters, separated by bridge plugs and cement plugs in 13⅜-in. casing inside a 17-in. wellbore in order to accommodate waste canisters with 10.75 in. (0.27 m) outside diameter over a depth interval from approximately 3,000 to 5,000 m. The total number of waste canisters emplaced in the 3- to 5-km interval is 400, assuming about 350 pressurized water reactor fuel rods per canister, which requires disassembling the fuel rod assemblies to achieve this density of rods in a canister (Arnold et al. 2011), for a total of about 250 MTHM per borehole (Arnold et al. 2011). This approach limits the mechanical stresses on the lower canisters from the weight of the overlying canisters and provides a degree of isolation for each canister string. An oil-based fluid with bentonite would be used in the waste disposal zone for emplacement of the canister strings. The borehole would be sealed using a series of compacted bentonite seals, bridge plugs, cement plugs, and backfill. The seals and plugs would be seated against the borehole wall from a depth of about 1,500 to 2,900 m. Large diameter casing (36-in. and 28-in.) that has been cemented into the borehole would be left in the upper 1,500 m of the borehole and the casing would be sealed with bridge plugs, cement plugs, and sand/crushed rock backfill (Arnold et al. 2011). [Note that 13⅜-in. slotted guidance casing in the waste emplacement zone from 3,000–5,000 m is left in place after waste emplacement (but it is not cemented to the wellbore), while the upper portion of the 13⅜-in. case from 3,000 m to the surface is cut and removed.]

4.2.3.2.4.2 Generic Deep Borehole Evolution

After closure, thermal effects from decay of the disposed high-activity waste would initially drive the evolution of a deep borehole repository. Vertical fluid flow in the borehole and surrounding DRZ would be driven by the thermal gradient arising from this decay heat. As described by Brady et al. (2009), temperatures at the borehole wall will peak relatively quickly (within about ten years of waste emplacement) at about 30°C higher than the ambient temperature of the host rock, assuming a single pressurized water reactor assembly aged for 25 years. This represents only a little more than half the number of fuel rods per disposal canister compared to the design described in Arnold et al. (2011). However, this temperature peak is highly dependent on the aging and fuel isotopic content (i.e., burn-up); for example, reprocessed commercial used nuclear fuel aged for 10 years showed a 150°C increase above ambient host rock temperatures (Brady et al. 2009). The heat generated from the waste will not only cause the fluid temperature to rise in the vicinity of the waste, but also the fluid pressure. The elevated pressure will drive fluid away from the heated zone, and the path of least resistance will be up the sealed borehole and adjacent disturbed zone, where permeabilities are likely to be higher than that of the undisturbed bedrock. The model results indicate that upward fluid flow in the heated borehole only persists for a relatively short period of time (<1,000 years) after emplacement. Fluid movement is primarily caused by the local elevated pressures due to thermal expansion of the pore water. As the heat generation decreases, the temperature of the waste decreases and the fluid begins to contract, lowering pressure. Buoyancy forces are not significant in this system because heat flow is primarily conductive rather than advective. The permeability of the sealed borehole would have to be significantly higher and there would have to be a source of water connected to the borehole by a high-permeability conduit in order for buoyancy-driven flow (i.e., a chimney effect) to be an important factor. Because the actual pore water density will likely increase with depth due to salinity stratification, the assumed conditions probably represent an upper bound on the fluid flow rates.

Both Brady et al. (2009) and Clayton et al. (2011) have conducted a preliminary safety assessment of a deep borehole repository that assumed transport of released radionuclides vertically upward through both the borehole and the disturbed zone annulus. The radionuclides must migrate through some portion of the 2-km waste disposal zone at the bottom of the borehole, followed by migration through the 1-km bentonite seal zone above the disposal zone, before encountering a region of the upper borehole that is assumed to intersect a potable aquifer. The driving force for the upward hydrologic gradient from the source (waste disposal) zone occurs for only 200 years, corresponding to the duration of the thermally driven flow. Subsequent to the thermal period, ambient conditions are not expected to provide any upward gradient, and upward radionuclide transport is assumed to cease.

Because the period of thermally driven flow (200 years) is short relative to the hydrologic travel time up the sealed borehole (~2,000 years), the only radionuclide with a non-zero concentration 1,000 m above the waste disposal zone in the sealed borehole is ^{129}I, which is assumed to have no retardation. The non-zero ^{129}I concentration represents the leading edge of the dispersive transport front. However, the center of mass never reaches the top of the 1,000 m sealed section of the borehole because there is no further movement after 200 years.

Human intrusion directly into the waste disposal sections of a deep borehole is very unlikely because the waste is located in the lower 2 km of a borehole with the upper 3 km sealed with concrete and backfill. Depending on site dependent characteristics such as resource availability and future drilling rate estimates, human intrusion is likely to be excluded from the analysis.

In summary, Brady et al. (2009) conclude that the vast majority of radionuclides, and the fuel itself, will be thermodynamically stable and will therefore resist dissolution into borehole fluids, or movement into and through the adjacent rocks. Vertical transport velocities during the early upward flow window will be low enough that total vertical fluid movement in, and adjacent to, deep borehole disposal zones should not exceed roughly 100 meters during this time. In the absence of advection, chemical diffusion cannot move radionuclides from boreholes through discontinuous, stagnant, and density-stratified waters over distances much greater than about 200 meters in the 1,000,000 years needed for the vast bulk of the radioactivity to decay away. Thus, simplified safety assessment calculations indicate that radiological dose to a human receptor via the vertical groundwater pathway up the borehole would be limited to a single radionuclide (^{129}I) and would be negligibly small, ~10s order of magnitude below current criteria.

4.2.3.3 Scenario Screening for the Generic Safety Case

Scenarios can be screened using similar criteria to those used for the screening of FEPs, for example the same regulatory, probability, and consequence criteria defined in Section 4.2.2.1.2. However, scenario screening requires site-, design-, and regulation-specific information. This is particularly true for assigning scenario probabilities, which are needed to evaluate the likelihoods of the scenarios. As a result, a formal scenario screening cannot be done generically. Instead, an informal scenario screening was performed to identify generic baseline scenarios for each of the four disposal options for implementation in simplified safety assessment models. As described below, these simplified baseline scenarios do not necessarily represent all aspects of expected or nominal conditions for each disposal option, but they do provide a consistent basis for preliminary generic disposal system evaluations and sensitivity analyses.

The following three scenario classes were identified as being of interest for the four generic geologic disposal options:

1. Undisturbed Scenarios
   a. Pathway 1: Liquid Phase Advective Transport
   b. Pathway 2: Liquid Phase Diffusive Transport
   c. Pathway 3: Gas Phase Transport
2. Defective EBS Scenarios
   a. Defective Waste Package
   b. Defective Backfill/Buffer
   c. Defective Sealing System
3. Disturbed Scenarios
   a. Human Intrusion
   b. Seismic Activity
   c. Igneous Event
   d. Glaciation

These scenario classes capture simplified high-level characteristics of (a) the descriptions of the initial state and the evolution to the final state in Section 4.2.3.2, and (b) the scenarios of interest in repository programs in other countries (Section 3.6 and Appendix C).

The undisturbed scenario class includes three release pathways. Multiple release pathways may be present in a single disposal system, and different release pathways may be dominant across different domains within a single disposal system.

The defective EBS scenario class includes three possible scenarios. Where these engineered components are present in a specific disposal system design, enhanced failure and/or degradation of these components may be included in the undisturbed scenario, thus eliminating the need for consideration of explicit defective EBS scenarios.

The four disturbed scenarios are caused by external events. Often during site selection, some of these disturbed scenarios may be excluded. For example, avoiding seismically-active or volcanically-active locations can reduce the probability of occurrence of these scenarios to below regulatory thresholds.

At this stage in the U.S. disposal program the focus is on generic options and feasibility. Disturbed scenarios, including human intrusion, are highly dependent on site specific information and regulatory considerations. Thus, evaluations of all but the undisturbed scenario are deferred in this Generic Safety Case at this stage. In some cases, defective EBS scenarios are incorporated into the undisturbed scenario. In the following subsections, details of these selected scenarios are summarized. Implementation of the selected scenarios in safety assessment models is described in Section 4.3.

The selected scenarios were constructed to represent a simplified baseline set of conditions for each disposal option, with some consistent conditions across disposal options. Each baseline scenario shares the following characteristics:

- Undisturbed conditions with the potential for advective and diffusive aqueous-phase transport. Gas-phase transport is not considered. Details of transport pathways for each disposal option are described in the following subsections.
- Defective waste packages that are assumed to fail instantaneously (i.e., at the beginning of the postclosure period)
- A radionuclide inventory that consists entirely of commercial used nuclear fuel, specifically pressurized water reactor fuel with a burn-up of 60 GWd/MTHM and 4.73% enrichment aged 30 years after discharge from a reactor.
- A hypothetical reference biosphere based on the International Atomic Energy Agency Example Reference Biosphere (ERB) 1B model (IAEA 2003b, Sections A.3.2 and C.2.6.1). The ERB 1B dose model assumes that the receptor is an individual adult who obtains his drinking water from a hypothetical drinking well drilled into the far field of the natural system. The ERB 1B model is used to convert the dissolved radionuclide concentrations in groundwater at the hypothetical drinking well location to an estimate of annual dose to a receptor based on the well dilution/pumping rate (assumed to be 10,000 m^3/yr), individual water consumption rate (1.2 m^3/yr), and radionuclide-specific dose conversion factors. The drinking well is assumed to capture all radionuclides transported out of the far field.

The selected baseline scenarios do not necessarily represent all aspects of expected or nominal conditions for each disposal option (e.g., instantaneous waste package failure in a granite repository), but they do provide a consistent basis for preliminary generic disposal system evaluations and sensitivity analyses. The baseline scenarios also provide a convenient starting point for future formal FEP analysis, scenario development, and safety assessment modeling as site- and/or design-specific information becomes available.

4.2.3.3.1 Generic Salt Repository Baseline Scenario

The description of the initial state and expected evolution of a generic salt repository in Section 4.2.3.2.1 identified diffusion as the only mechanism for radionuclide transport away from a salt repository under undisturbed conditions. The salt baseline scenario includes radionuclide transport pathways based on the WIPP (Appendix C, Section C-3.1.2). Specific baseline scenario transport pathways include:

- **Near field**—Vertical diffusive transport through a 5-m thickness of (a) creep consolidated crushed salt backfill, and (b) the disturbed rock zone halite between the excavation and an underlying anhydrite interbed. Chemically reducing conditions, consistent with a concentrated brine.
- **Far field**—Horizontal transport through an underlying anhydrite interbed. Brine flow in the interbed is assumed to vary over a range of flow rates. At the high end of the range, advective transport can occur. The drinking water well (i.e., the receptor location) intersects the interbed at a distance of 5,000 m from the repository. Less chemically reducing conditions than in the near field, consistent with a more dilute brine.

The focus here is on bedded salt but the baseline can be extended to domal salt formations. One area where there is a difference is that a domal formation would not contain interbed pathways.

4.2.3.3.2 Generic Clay Repository Baseline Scenario

The description of the initial state and expected evolution of a generic clay repository in Section 4.2.3.2.2 identified the following radionuclide transport pathways:

Undisturbed Scenario Pathways:

- **Advective Transport through the Disturbed Rock Zone and Shaft Seals**—Fluid flow through the repository openings or DRZ and up the shafts transports radionuclides to a shallow aquifer from which they are pumped to the biosphere. This pathway scenario requires sufficiently high permeability within the repository, the DRZ, and seals, and a sustained upward hydraulic gradient. An upward gradient could result from: (1) ambient hydrologic conditions, (2) thermal pressurization of fluid within the waste disposal zone from waste heat, (3) buoyancy of heated fluid within the waste disposal zone, or (4) thermo-chemical reactions that release water and/or gases within the waste disposal zone.

  Only ambient hydrologic conditions would operate over long timescales; the thermally driven mechanisms would cease within a few hundred years. Self-sealing of the DRZ, and eventual consolidation of clay components in the shaft and emplacement borehole seals, would limit such a high permeability pathway. Finally, advection along the DRZ and up the access shafts requires a source of water inflow from the clay with sufficient strength to result in water saturation above residual saturation so there is a continuous water phase present, and with greater hydraulic head than the overlying aquifer to maintain flow. Such a pressure drive could result if sufficient gas generation results from corrosion, radiolysis, or biodegradation.
- **Diffusive Transport in Host Clay**—Diffusion transports radionuclides upward from the repository, through the clay host rock, to a shallow aquifer from which they are pumped to the biosphere. Advective transport (under realistic hydraulic gradients) is likely to be modest during the repository performance period, given the low permeability of clay media. The chemically reducing environment typical for clays and shales would limit the solubility of most radionuclides, thus limiting mobility. Also, minerals present in clay formations readily sorb many radionuclides, further attenuating releases. Finally, because of the long- transport times, radioactive decay can be an important process.

The clay baseline scenario considers vertical diffusive transport through a 150-m thickness of host clay. The drinking water well intersects an aquifer immediately above the host clay. The advective transport pathway through the DRZ and shaft seals is screened out due to the low probability of maintaining (1) an ambient upward hydraulic gradient over long timescales, and (2) a high permeability in the DRZ and seals as they consolidate over time.

4.2.3.3.3 Generic Granite Repository Baseline Scenario

The description of the initial state and expected evolution of a generic granite repository in Section 4.2.3.3.2 identified the following radionuclide transport pathways:

Undisturbed Scenario Pathways—Waste packages and engineered barriers are assumed to perform as designed. No releases are likely because the waste form remains contained within the waste package during the entire performance period (Mariner et al. 2011). This scenario maintains that in the absence of external events, the buffer will protect the waste packages from significant damage during the performance period, preventing canister failure and radionuclide release. This scenario is consistent with the base scenario of the third case study for the Canadian concept (Gierszewski et al. 2004) and the expected performance of the Swedish repository, barring an unexpectedly large earthquake or significant buffer erosion (SKB 2006b).

Defective EBS Scenario Pathways:

- **A Major Defect in a Waste Package Allows Early Radionuclide Release**—One or more waste packages are assumed to have a major defect at the time of emplacement. The number of waste packages experiencing defects is determined by specific waste packaging designs and manufacturing and emplacement methods. The release of radionuclides under such a scenario also depends on specifics of the waste form. For example, used fuel cladding would have to also fail in the defective waste package. This scenario assumes that both the cladding and the waste package fail. The consequences of a waste package defect are bounded by assuming the failed waste package provides no barrier performance. Under this scenario, radiolytic phenomena may increase waste form degradation rates above solubility limited rates. The buffer, backfill, and seals perform as designed, causing the primary pathway to be through the geosphere. Released radionuclides diffuse through the bentonite buffer and migrate to the biosphere via far-field fractures.
- **Buffer Failure Scenario**—Corrosion of a number of waste canisters is enhanced by the erosion of buffer materials caused by hydrologic changes brought on by the next glacial cycle. The consequences from such a scenario are site specific and depend on the frequency, duration, and intensity of glacial cycles at the site, which will affect the amount and degree of buffer erosion. Some assumptions can be obtained from an SKB analysis (SKB 2006b) where it is assumed that the earth's glacial cycle continues such that an ice sheet or glacier advances over the top of the repository site and then, at approximately 100,000 years in the future, retreats during a subsequent warming period. The warming period causes deep penetration of melt waters at the repository site as the ice retreats. The increased flow conditions at depth last approximately 25,000 years and erode the buffer to expose some of the waste packages to advective groundwater flow. The increased corrosion rate due to flowing groundwater causes a small fraction of these waste packages to fail within 1,000,000 years. Internal waste package components, such as the used fuel cladding, are pessimistically assumed to fail when the canister fails, initiating degradation of all waste in the failed packages. Thus, radionuclides released from breached waste packages are assumed to migrate directly to the host rock.

The granite baseline scenario includes horizontal transport through a 0.78-m near-field pathway (bentonite buffer and granite DRZ) and a 5,000-m far-field pathway (granite fractures and matrix) under chemically reducing conditions. The drinking water well intersects the fractured granite at a distance of 5,000 m from the repository. The baseline scenario also includes the effects of defective waste packages; 1% of the waste packages (and cladding) are assumed to fail instantaneously, the other 99% are assumed to remain intact for 10,000,000 years. This differs from the other baseline scenarios, where all waste packages are assumed to fail instantaneously, to reflect the expected longevity of waste packages and the buffer in a granite repository (Section 4.2.3.2.3.2). The buffer failure scenario is screened out of this generic baseline scenario because it is a site-specific phenomenon.

4.2.3.3.4 Generic Deep Borehole Disposal Baseline Scenario

The description of the initial state and expected evolution of a generic deep borehole disposal facility in Section 4.2.3.2.4 identified the following radionuclide transport pathways:

Undisturbed Scenario Pathways:

- **Transport in the Borehole**—Fluid flow up the borehole transports radionuclides to a shallow aquifer from which they are pumped to the biosphere. This scenario requires sufficiently high permeability within the borehole and a sustained upward gradient in hydrologic potential for it to occur. Vertical permeability within the borehole in the waste disposal zone may be relatively high. Rapid degradation of the disposal canisters stacked within the borehole is assumed. Vertical permeability within the borehole above the level of waste emplacement will be engineered to be very low, significantly reducing fluid flow and creating diffusion-dominated transport conditions in this portion of the borehole. Some upward gradient in hydrologic potential (i.e., advection) within the borehole could result from (a) ambient hydrologic conditions, (b) thermal pressurization of fluid within the waste disposal zone from waste heat, (c) buoyancy of heated fluid within the waste disposal zone, or (d) thermo-chemical reactions that release water and/or gases within the waste disposal zone. The duration of the thermal pulse is small compared to the regulatory period and occurs during the time when the upper sealing system is likely to be the most robust.
- **Transport in Disturbed Rock around the Borehole**—Fluid flow up the annulus of disturbed rock surrounding the borehole transports radionuclides to a shallow aquifer from which they are pumped to the biosphere. This scenario requires sufficiently high permeability in the disturbed zone surrounding the borehole and a sustained upward gradient in hydrologic potential for it to occur. Vertical permeability within disturbed rock in the waste disposal zone and in the overlying rock may be higher than that of the surrounding intact rock or intact sealing system components. Vertical permeability in the crystalline rock immediately outside the heated volume near the waste disposal zone could be increased because thermo-mechanical effects would reduce the vertical mechanical stress. An upward gradient in hydrologic potential within the annulus of the borehole could result from (a) ambient hydrologic conditions, (b) thermal pressurization of fluids within the waste disposal zone from waste heat, (c) buoyancy of heated fluids within the waste disposal zone, or (d) thermo-chemical reactions that release water and/or gases within the waste disposal zone.
- **Transport in Surrounding Rock Away from the Borehole**—Fluid flow up through the crystalline basement and sedimentary cover transports radionuclides to a shallow aquifer from which they are pumped to the biosphere. This scenario requires sufficiently high permeability within fracture zones and/or faults in the crystalline basement and sedimentary cover and a sustained upward gradient in hydrologic potential for it to occur. Given the low vertical permeability of the crystalline basement rocks and the stratified sedimentary cover, a through-going feature such as an interconnected group of fracture zones or faults would be required to conduct significant quantities of fluid to a shallow aquifer.

Defective EBS Scenario Pathways: These scenarios have the same three pathways as the undisturbed scenario but the consequences conditional on failed seals and backfill are likely to be larger because of their condition. These include: failed borehole sealing in the upper 3 km of the borehole, and failed or improper grouting during borehole construction and/or abandonment. This would result in increased vertical flow through the borehole and DRZ as well as increased lateral connectivity between the borehole and surrounding intact host rock.

The deep borehole baseline scenario combines the first two undisturbed transport pathways: up the borehole; and up the DRZ around the borehole. The transport pathway includes a 2,000-m waste disposal zone and a 1,000-m seal zone under chemically reducing conditions. The drinking water well is assumed to intersect the borehole upper zone (spanning from 0 – 2,000 m depth). The third undisturbed transport pathway, into the surrounding rock away from the borehole, is screened out due to the low permeability of basement crystalline rock relative to the borehole pathways and the low probability of a continuous 3,000-to-5,000-m fracture or fault from the deep basement to a hypothetical overlying aquifer. The defective EBS scenarios are also screened out. Their effects are partially accounted for in the characterization of the DRZ and in sensitivity analyses.

4.3 Generic Disposal System Model Development

4.3.1 Introduction

A postclosure safety assessment is performed to provide a quantitative estimation of the behavior of a geologic disposal system and its component subsystems. Within the safety assessment methodology described in Section 2.3.1, the quantitative safety assessment includes the following steps (as numbered in Section 2.3.1):

4. Build models and abstractions
   a. Conceptual models
   b. Mathematical models
   c. Computational models
5. Quantify uncertainty
6. Construct integrated safety assessment model and perform calculations
7. Perform uncertainty and sensitivity analyses
8. Evaluate performance

To support this Generic Safety Case, simplified postclosure safety assessments were performed for each of the four geologic disposal options outlined in Section 1.4: mined disposal in salt, clay, and granite formations; and deep borehole disposal in crystalline rock. The remainder of Section 4.3 describes the conceptual, mathematical, and computational models used to represent the selected baseline scenarios (from Section 4.2.3.3) for each of the four disposal options. Section 4.4 presents and discusses the safety assessment model results for each of the disposal options, including sensitivity analyses. Section 5.2.3 provides a synthesis of the results in the context of the generic safety case.

The four safety assessment models were developed within a common conceptual framework, organized around four common disposal system regions (Figure 2-1): Source; Near Field; Far Field; and Biosphere. Each of the four regions, in turn, consists of one or more common features. These regions and features are consistent with the generic disposal system regions shown in Figure 4-1 supporting the FEP identification and classification. Within the common conceptual framework, representations of the system components and relevant phenomena (i.e., the included FEPs and scenarios) can range from simple abstractions to complex coupled processes.

To support this Generic Safety Case, the safety assessment models focus on the generic baseline scenarios for each of the four disposal options described in Section 4.2.3.3. Each of the four baseline scenarios includes both advective and diffusive aqueous-phase transport, although one or the other may be dominant in specific regions depending on the engineered and natural barrier properties. The baseline scenarios also include the effects of the defective waste packages, which results in the early onset of waste form degradation and early-time radionuclide release.

In site-specific safety assessments, the disturbed scenarios – human intrusion, seismic, and igneous – often provide the dominant radionuclide releases contributing to dose. However, as noted in Section 4.2.3.3, the relative importance of the disturbed scenarios can be minimized through site selection. Therefore, for the four generic safety assessments presented here, the potential consequences of the disturbed scenarios are not explicitly evaluated at this time. As site-specific considerations become important in future safety assessments, the impact of disturbed scenarios to each of the four disposal options will be re-evaluated as appropriate.

For these generic safety assessments, simple mathematical representations of the FEPs (e.g., reduced-dimension geometry, minimal multi-physics process coupling) were used. These simple representations are sufficient to perform high-level generic safety calculations of the four disposal system baseline scenarios and to demonstrate the process of safety assessment evaluation. The simplified mathematical models used to represent the source term, radionuclide transport in the near field and far field, and radionuclide doses are presented in Clayton et al. (2011, Section 4.1.1). The conceptual framework and mathematical representations were implemented within the GoldSim computational framework (GoldSim Technology Group 2010a). The GoldSim Contaminant Transport Module (GoldSim Technology Group 2010b) provides numerical solutions to the simple mathematical models.

4.3.2 Safety Assessment Model Descriptions

In Fiscal Year (FY) 2011, simplified generic disposal system (GDS) models were developed and applied to generic salt, clay, granite and deep borehole disposal options (Clayton et al. 2011, Section 3). These four models, referred to as the FY 2011 GDS models, provide the basis for the four GDS safety assessment baseline scenario models developed to support this Generic Safety Case. Specific details of each of the safety assessment baseline scenario models are summarized in the following subsections, including the identification of input parameter values.

An important difference from the modeling approach used for the FY 2011 GDS models (Clayton et al. 2011, Section 3) is in the treatment of uncertainty. In the FY 2011 GDS simulations, uncertain parameters were defined by a parameter distribution. Parameter uncertainty was propagated into the models by conducting multiple realizations for each scenario; for each realization a different set of values for the uncertain parameters was sampled from the distributions of possible values (e.g., Monte Carlo simulation). In a statistical sense, each individual realization represents a different possible representation of the future overall performance of the system, consistent with the uncertainty in the input parameters.

In the Generic Safety Case simulations described here, each baseline scenario is modeled with a single deterministic simulation, where uncertain parameters are represented by their mean value. The effects of parameter uncertainty are examined by performing “one-off” simulations, where an uncertain parameter is varied from its mean value. Uncertainty treatment is described in more detail in Sections 2.3.1 (Step 6) and 5.2.

For each GDS safety assessment baseline scenario, the following deterministic simulations were performed:

- **Baseline Analysis—**A single deterministic simulation of the baseline scenario. Each uncertain parameter (i.e., those parameters defined by a distribution in the probabilistic models) was represented by its mean value. Baseline analysis results are presented in Section 4.4.1.
- **Sensitivity Analyses**—A set of “one-off” deterministic simulations. A single uncertain parameter value was varied from the baseline mean value in each simulation. Sensitivity analysis results are presented in Section 4.4.2.

In each of the following subsections the GDS safety assessment baseline model descriptions include modifications that were made to the FY 2011 GDS models for the deterministic application and to provide increased consistency across the four disposal options. In addition, the following other common changes were made to the FY 2011 GDS models; these changes resulted in more consistency across the four GDS safety assessment baseline scenarios used to support the safety case:

- The waste inventory for each of the three mined disposal options assumed a repository capacity of 70,000 MTHM. The entire 70,000 MTHM radionuclide inventory was assumed to be commercial UNF, specifically pressurized water reactor fuel with a burn-up of 60 GWd/MTHM and 4.73% enrichment aged 30 years after discharge from a reactor (Carter and Luptak 2010, Table C-1). The 70,000 MTHM UNF inventory was assumed to be contained in 16,000 waste packages, with each waste package containing 10 pressurized water reactor assemblies. For deep borehole disposal the repository capacity affects the total number of boreholes required (approximately 400 boreholes would be required to dispose of 70,000 MTHM), but does not affect the conceptualization of an individual borehole. A discussion of the radionuclide makeup of 70,000 MTHM model inventory is presented in Appendix E.
- The fractional waste form degradation rate for each of the four disposal options was assumed to be 2×10^{-5} yr^{-1}, which is the most likely (mode) value from the FY 2011 clay GDS model (Clayton et al. 2011, Section 3.3.3.3.2). At this fractional rate, 50% of the radionuclide mass is released from the waste form in the first 35,000 years, 95% of the mass is released by 150,000 years, and 99.9% of the mass is released by about 350,000 years. A slower fractional degradation rate of 1×10^{-7} yr^{-1}, consistent with the FY 2011 salt, granite and deep borehole GDS models, was examined as part of the sensitivity analyses.
- Simulations were run to 10,000,000 years and used consistent time stepping for the purpose of investigating performance out to peak dose.

4.3.2.1 Deterministic Salt GDS Model

The deterministic salt GDS safety assessment model supporting the safety case derives from the FY 2011 salt GDS model (Clayton et al. 2011, Section 3.1). The salt baseline scenario, described in Section 4.2.3.3.1, includes transport through the near-field (creep consolidated backfill and salt DRZ) and far-field (anhydrite interbed) pathways. The baseline scenario does not attribute any barrier capability to the waste packages; they are assumed to fail instantaneously. The baseline scenario also does not attribute any sorptive capacity to the corrosion products or backfill.

Changes from the FY 2011 salt GDS model (Clayton et al. 2011, Section 3.1.4.1.1) to support the safety case include the following:

- Deterministic 10,000,000-year simulation with mean values for uncertain parameters
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Reduced repository length from 3,270 m to 2,146 m to be consistent with the smaller number of waste packages
- Brine flow rates through the near-field salt DRZ and far-field interbed assumed to remain constant at the 1,000,000-year value until 10,000,000 years.

Additional details describing the deterministic salt GDS model input parameters are presented in Appendix E. The resulting salt baseline scenario model supporting the safety case is summarized in Table 4-4.

Simulation results from the deterministic salt baseline scenario are presented in Section 4.4.1.1. Results from one-off sensitivity simulations are presented in Section 4.4.2.1.

It should be noted that the salt baseline scenario simulated here is representative of bedded salt. For a domal salt scenario, the far-field host salt would have properties of intact halite rather than an anhydrite interbed, but would not extend as far as the interbed.

Table 4-4. Summary of the Salt Baseline Scenario Model

Region	Feature	Salt Model	Baseline Scenario Representation
Source	Inventory	Used Nuclear Fuel	70,000 MTHM
	Waste Form	Used Nuclear Fuel	2×10^{-5} yr^{-1} fractional degradation rate, no cladding credit
Near Field	Waste Package	Waste Package	Instantaneous failure
	Buffer / Backfill	Included in DRZ	Not applicable
	Seals / Liner	Not modeled	Not applicable
	DRZ	Near-Field Salt (5 m)	Diffusive transport, no sorption
Far Field	Host Rock	Salt Interbed (5,000 m)	Diffusive transport with sorption
	Aquifer	IAEA BIOMASS ERB1B	10,000 m^3/yr dilution rate
Receptor	Surface / Biosphere	IAEA BIOMASS ERB1B	1.2 m^3/yr water consumption rate ERB 1 Dose Coefficients

4.3.2.2 Deterministic Clay GDS Model

The deterministic clay GDS safety assessment model supporting the safety case derives from the FY 2011 clay GDS model (Clayton et al. 2011, Section 3.3). The clay baseline scenario represents the host clay diffusive transport pathway identified in Section 4.2.3.3.2, which includes transport through near field (bentonite buffer and clay DRZ) and far field (host clay). For this safety assessment, the clay baseline scenario does not attribute any barrier capability to the waste packages; they are assumed to fail instantaneously.

Changes from the FY 2011 clay GDS model (Clayton et al. 2011, Section 3.3.4.2.1) to support the safety case include the following:

- Deterministic 10,000,000-year simulation with mean values for uncertain parameters
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Instantaneous waste package failure
- Clay thickness of 150 m overlying the emplaced waste, consistent with Hansen et al. (2010, Figure 2.1-1 and Section 4)
- Equivalent diffusive releases to the far-field clay in both the upward and downward directions

Additional details describing the deterministic clay GDS model input parameters are presented in Appendix E. The resulting clay baseline scenario model supporting the safety case is summarized in Table 4-5.

Simulation results from the deterministic clay baseline scenario are presented in Section 4.4.1.2. Results from one-off sensitivity simulations are presented in Section 4.4.2.2.

Table 4-5. Summary of the Clay Baseline Scenario Model

Region	Feature	Clay Model	Baseline Scenario Representation
Source	Inventory	Used Nuclear Fuel	70,000 MTHM
	Waste Form	Used Nuclear Fuel	2×10^{-5} yr^{-1} fractional degradation rate, no cladding credit
Near Field	Waste Package	Waste Package	Instantaneous failure
	Buffer / Backfill	Bentonite (1.025 m)	Diffusive transport with sorption
	Seals / Liner	Not modeled	Not applicable
	DRZ	Fissured Clay (1.15 m)	Diffusive transport with sorption
Far Field	Host Rock	Clay (150 m)	Diffusive transport with sorption
	Aquifer	IAEA BIOMASS ERB1B	10,000 m^3/yr dilution rate
Receptor	Surface / Biosphere	IAEA BIOMASS ERB1B	1.2 m^3/yr water consumption rate ERB 1 Dose Coefficients

4.3.2.3 Deterministic Granite GDS Model

The deterministic granite GDS safety assessment model supporting the safety case derives from the FY 2011 granite GDS model (Clayton et al. 2011, Section 3.2). The granite baseline scenario, described in Section 4.2.3.3.3, includes transport through the near-field (bentonite buffer and granite DRZ) and far-field (fractured granite) pathways. For this safety assessment, the granite baseline scenario includes the effects of defective waste packages; 1% of the waste packages (and cladding) are assumed to fail instantaneously.

Changes from the FY 2011 granite GDS model (Clayton et al. 2011, Section 3.2.3.2.1) to support the safety case include:

- Deterministic 10,000,000-year simulation with mean values for uncertain parameters
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Replace the three-dimensional representation of far-field fractured granite using the FEHM dynamically-linked library with a one-dimensional GoldSim pipe with matrix diffusion.
- Replace the two-dimensional representation of bentonite buffer with a set of one-dimensional GoldSim cells
- Update solubility values to be more representative of granite pore waters (based on Mariner et al. 2011, Table 2-5)
- Update distribution coefficients (k_d's) to be more representative of bentonite in the waste package and buffer, based on the waste package and bentonite k_d values used in the clay GDS model (Table E-2 and Clayton et al. 2011, Section 3.3.3.3)
- Update distribution coefficients (k_d's) to be more representative of granite in the host rock (based on Carbol and Engkvist 1997).

- Instantaneous failure of 1% (160) of the waste packages. This replaces the FY 2011 GDS model assumption that between 0.1% and 1% of the waste packages directly intersect a far-field fracture.

Additional details describing the deterministic granite GDS model input parameters are presented in Appendix E. The resulting granite baseline scenario model supporting the safety case is summarized in Table 4-6.

Simulation results from the deterministic granite baseline scenario are presented in Section 4.4.1.3. Results from one-off sensitivity simulations are presented in Section 4.4.2.3.

Table 4-6. Summary of the Granite Baseline Scenario Model

Region	Feature	Granite Model	Baseline Scenario Representation
Source	Inventory	Used Nuclear Fuel	70,000 MTHM
	Waste Form	Used Nuclear Fuel	2×10^{-5} yr^{-1} fractional degradation rate, no cladding credit
Near Field	Waste Package	Waste Package	Instantaneous failure of 1% of waste packages
	Buffer / Backfill	Bentonite (0.36 m)	Diffusive transport with sorption
	Seals / Liner	Not modeled	Not applicable
	DRZ	Granite (0.42 m)	Advective transport with sorption
Far Field	Host Rock	Granite (5,000 m)	Advective transport in fractures, with sorption and matrix diffusion
	Aquifer	IAEA BIOMASS ERB1B	10,000 m^3/yr dilution rate
Receptor	Surface / Biosphere	IAEA BIOMASS ERB1B	1.2 m^3/yr water consumption rate ERB 1 Dose Coefficients

4.3.2.4 Deterministic Deep Borehole GDS Model

The deterministic deep borehole GDS safety assessment model supporting the safety case derives from the FY 2011 deep borehole GDS model (Clayton et al. 2011, Section 3.4). The deep borehole baseline scenario, described in Section 4.2.3.3.4, combines two transport pathways: up the borehole; and up the DRZ around the borehole. The baseline scenario does not attribute any barrier capability to the waste packages; they are assumed to fail instantaneously.

Changes from the FY 2011 deep borehole GDS model (Clayton et al. 2011, Section 3.4.2.2.1) to support the safety case include the following:

- Deterministic 10,000,000-year simulation with mean values for uncertain parameters
- Waste inventory of 174 MTHM UNF in 400 waste packages in a single borehole
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Fluid flow rates up the borehole assumed to remain constant at the 1,000,000-year values until 10,000,000 years

Additional details describing the deterministic deep borehole GDS model input parameters are presented in Appendix E. The resulting deep borehole baseline scenario model supporting the safety case is summarized in Table 4-7.

Simulation results from the deterministic deep borehole baseline scenario are presented in Section 4.4.1.4. Results from one-off sensitivity simulations are presented in Section 4.4.2.4.

Table 4-7. Summary of the Deep Borehole Baseline Scenario Model

Region	Feature	Deep Borehole Model	Baseline Scenario Representation
Source	Inventory	Used Nuclear Fuel	174 MTHM
	Waste Form	Used Nuclear Fuel	2×10^{-5} yr^{-1} fractional degradation rate, no cladding credit
Near Field	Waste Package	Waste Package	Instantaneous failure
	Buffer / Backfill	Disposal Zone Degraded Waste (2,000 m)	Advective transport with sorption
	Seals / Liner	Seal Zone Bentonite (1,000 m)	Diffusive transport with sorption
	DRZ	Included in Seals	Included in seals
Far Field	Host Rock	Not modeled	Not applicable
	Aquifer	Upper Borehole Rock Materials (2,000 m)	10,000 m^3/yr dilution rate
Receptor	Surface / Biosphere	IAEA BIOMASS ERB1B	1.2 m^3/yr water consumption rate ERB 1 Dose Coefficients

4.4 Generic Safety Assessment Model Results

This section presents the results from the application of the four deterministic safety assessment baseline models described in Section 4.3.2. As noted in Section 1.1, it is not the intention of this Generic Safety Case to identify, screen, and/or prioritize specific disposal options, designs, and sites for their suitability for a geologic disposal facility. As such, these initial simplified safety assessment results are not intended to provide absolute indications of dose or to provide for comparisons between disposal options. Rather, these results for four generic disposal options seek to provide confidence that used nuclear fuel and high-level radioactive waste can be disposed of safely in the U.S. in mined disposal facilities in salt, clay, and granite formations, in deep borehole disposal in crystalline rocks.

Safety assessment results for the four baseline scenarios are presented in Section 4.4.1. The baseline scenario results, from deterministic simulations, provide a preliminary indication of estimated dose, given the baseline assumptions about source term, near-field, far-field, and biosphere properties.

Sensitivity analyses are presented in Section 4.4.2. These sensitivity analyses are one-off deterministic simulations from the baseline simulations and provide preliminary insights into which parameters, features, and/or barriers most significantly contribute to the overall capability of a specific disposal system to isolate waste from the biosphere under baseline scenario conditions.

4.4.1 Baseline Scenario Analyses

4.4.1.1 Deterministic Salt GDS Model Results

The deterministic salt baseline scenario is summarized in Section 4.3.2.1 and Table 4-4. Input parameters are listed in Clayton et al. (2011, Section 3.1) and Table E-1. As described in Section 4.2.3.2.1.2, there is no credible mechanism for movement of radionuclides from a salt repository under undisturbed conditions, except by diffusion, which is extremely slow.

The salt baseline scenario assumes an undisturbed transport pathway, but takes only minimal credit for the engineered barrier system; 95% of waste form degradation occurs in 150,000 years, all waste packages fail instantaneously, and there is no sorption in the near-field salt DRZ between the repository and the underlying interbed. The dose receptor is located 5,000 m from the repository. The resulting deterministic annual dose over 10,000,000 years is shown in Figure 4-2.

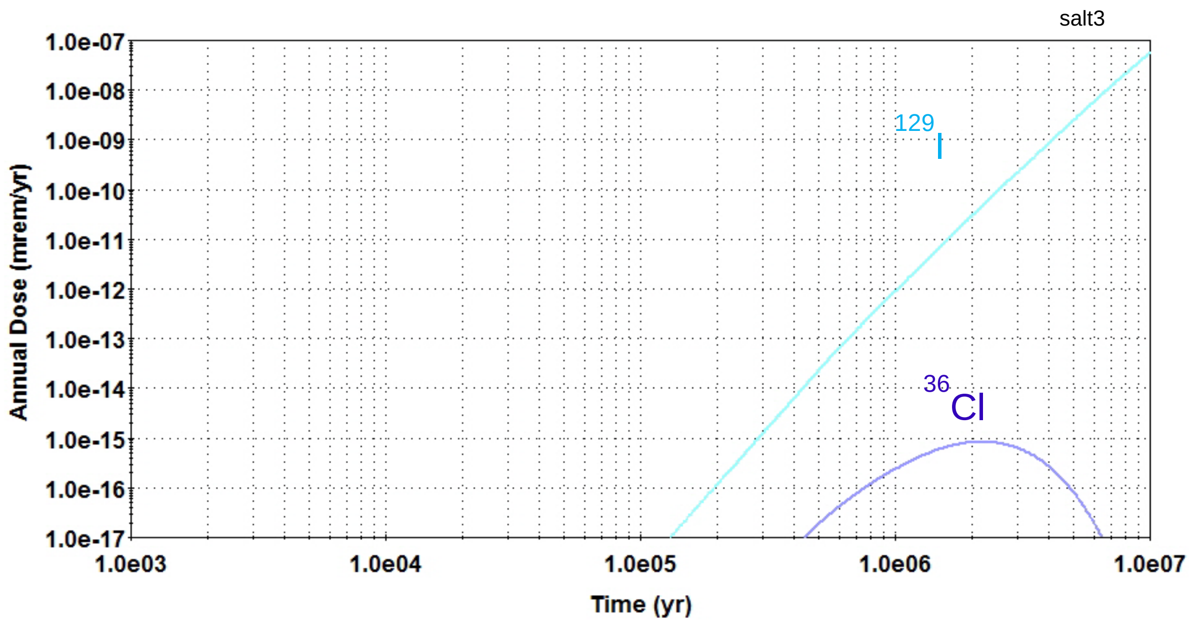


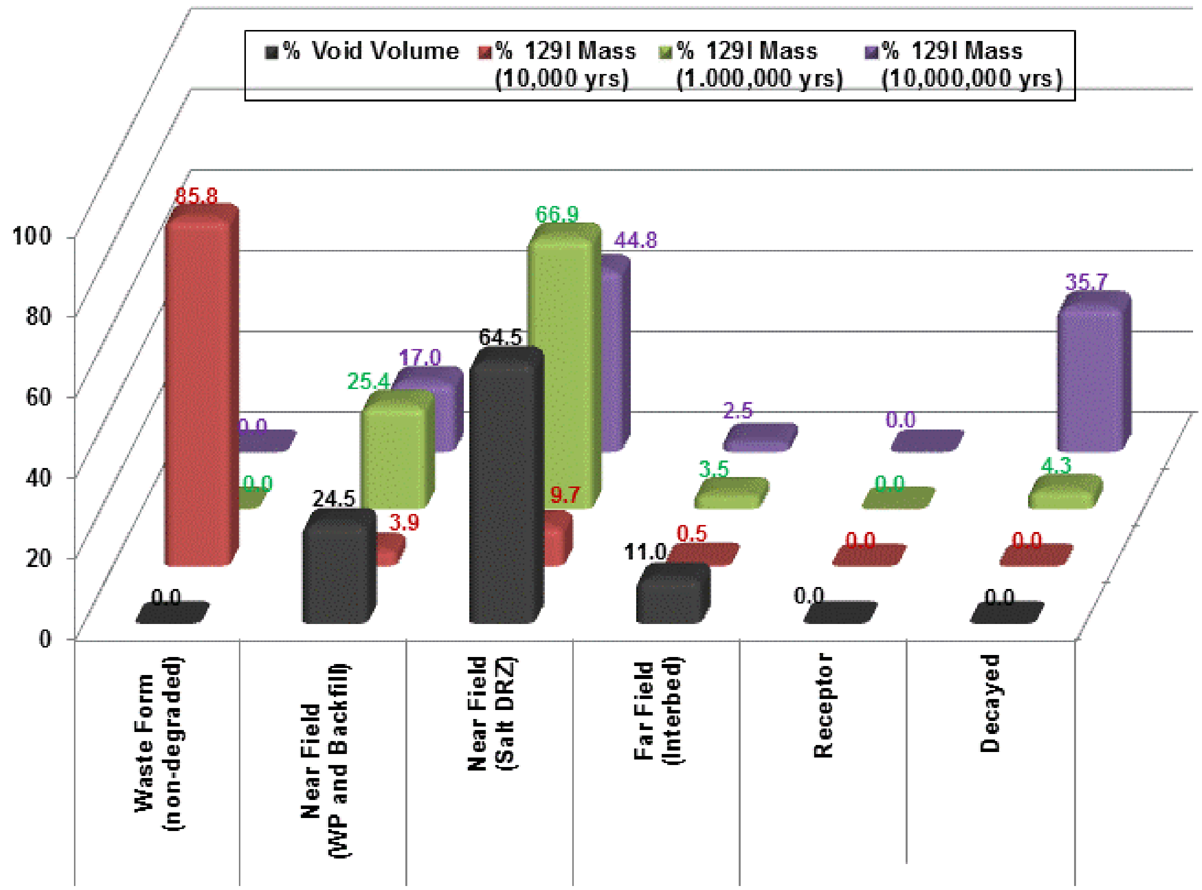
Figure 4-2. Salt Baseline Scenario Annual Dose for a Receptor 5,000 m from the Repository

In the first 10,000 years, the peak annual dose is < 1×10^{-23} mrem/yr; in the first 1,000,000 years, the peak annual dose is 9.0×10^{-13} mrem/yr (at 1,000,000 years). The peak annual dose over the entire 10,000,000-year simulation is 5.6×10^{-8} mrem/yr, occurring at 10,000,000 years. It should be noted that the salt baseline scenario does not consider a vertical diffusion pathway to the surface, a distance of several hundred meters as compared to the 5,000-m distance along the interbed to the receptor well. A vertical diffusion pathway, with subsequent surface transport to a receptor, could result in a somewhat higher peak annual dose than that shown in Figure 4-2 (see Section 4.4.2.1 for further discussion).

The baseline scenario dose is dominated by ^{129}I, with a minor contribution from ^{36}Cl. These are the only two radionuclides assumed to have no sorption (k_d = 0 mL/g) throughout the disposal system, unlimited solubility, and long half-lives – 15,700,000 years for ^{129}I and 301,000 years for ^{36}Cl. The larger initial mass (1,363 g/WP of ^{129}I as compared to 2.2 g/WP of ^{36}Cl) and longer half-life explain why the dose contribution is much larger from ^{129}I than from ^{36}Cl.

The deterministic safety case annual dose history in Figure 4-2 is similar to mean annual dose history estimated with the FY 2011 salt GDS probabilistic model (Clayton et al. 2011, Figure 3.1-8). Both dose histories are dominated by ^{129}I, with a minor contribution from ^{36}Cl. The deterministic annual dose supporting the safety case is generally about two orders of magnitude higher due to the faster waste form degradation rate (2×10^{-5} yr^{-1} vs. a range from 1×10^{-8} to 1×10^{-6} yr^{-1}). Lesser effects result from the shortened repository length (2,146 m vs. 3,270 m), which increases the dose, and the reduced inventory (~70,000 MTHM vs. ~90,000 MTHM), which decreases the dose. The sensitivity analyses in Section 4.4.2.1 further examine processes affecting generic salt disposal system performance.

The relative distribution of ^{129}I mass in the natural and engineered barriers of the salt disposal system at three different times during the deterministic simulation supporting the safety case is shown in Figure 4-3.



NOTE: % Void Volume provides an indication of the relative distribution of water volume across the components of the disposal system. The Waste Form, Receptor, and Decay components do not have any void volume.

Figure 4-3. Distribution of ^{129}I in the Salt GDS Model Components

At 10,000 years, 85.8% of the initial ^{129}I mass is still bound in the waste form. Less than 1% of the initial mass has been transported (by diffusion) beyond the 5-m thick near-field salt DRZ between the repository and the underlying interbed. At 1,000,000 years, the waste form has completely degraded, but only 3.5% of the initial ^{129}I mass has diffused beyond the near-field salt to the underlying interbed. The small dose at 1,000,000 years (9.0×10^{-13} mrem/yr) is due to a negligible mass of ^{129}I (2×10^{-8} g out of an initial repository ^{129}I mass of 21,830 kg) actually reaching the receptor. This negligible calculated mass at the receptor is effectively zero as it is of the same magnitude as the numerical precision of the solution. At 10,000,000 years, 35.7% of the initial ^{129}I mass has decayed. Most of the undecayed mass has still not diffused to the underlying interbed. Even at 10,000,000 years, only 0.01 g has reached the receptor.

Based on these deterministic salt GDS model results, the following observations can be made regarding the performance of a generic salt disposal system under baseline scenario conditions:

- Radionuclide releases to the receptor location in the biosphere are minimal; for long-lived non-sorbing ^{129}I, releases are effectively zero after 10,000,000 years. The peak dose is 5.6×10^{-8} mrem/yr at 10,000,000 years.
- Radionuclide transport through the near field (the engineered barrier system and the near-field salt DRZ between the repository and the underlying interbed) is slow due to:
  - Very low brine flow rates resulting in diffusion-dominated transport, and
  - Salt creep closure of the repository excavation and DRZ which minimizes the potential for high-permeability fracture connections to the underlying interbed.
- Radionuclide transport through the far field (anhydrite interbed) is slow due to:
  - Very low brine flow rates resulting in diffusion-dominated transport,
  - Radionuclide sorption,
  - Absence of well-connected fractures in the interbed, and
  - Long migration distance (5,000 m) to the receptor location.

The factors identified above as having a significant impact on near-field and far-field radionuclide transport are generally consistent with those of previous safety assessments of salt disposal systems, as discussed in Section 3.6 and Appendix C-3.1. For example, the studies in the United States (generic and WIPP) and Germany (Gorleben) also point to very low flow rates and the self-healing nature of salt as contributing to barrier performance.

Additional characteristics of a generic salt disposal system which were not captured in the deterministic salt GDS baseline scenario model include:

- Enhanced performance due to:
  - Radionuclide sorption in the engineered barrier system and near-field salt DRZ
  - Slower waste form degradation due to reducing chemical conditions
  - Thermally enhanced creep closure and dry out
  - Removing the model assumption that all radionuclides diffuse downward to the underlying interbed; the dose would be halved if upward diffusion were assumed and there were no equivalent overlying interbed.
- Degraded performance due to:
  - Instantaneous release of gap and grain boundary inventory from the waste form
  - Increased brine flow rates through the near-field DRZ and far-field interbed resulting from (a) well-connected fractures, and/or (b) repository pressurization from creep closure and gas generation
  - Reduced distance to the receptor location

Some of these characteristics are examined in the sensitivity analyses in Section 4.4.2.1.

4.4.1.2 Deterministic Clay GDS Model Results

The deterministic clay baseline scenario is summarized in Section 4.3.2.2 and Table 4-5. Input parameters are listed in Clayton et al. (2011, Section 3.3) and Table E-2. As described in Section 4.2.3.2.2.2, radionuclide releases from a clay repository disposal horizon would be limited by low advection, the reducing chemical environment, and sorption under undisturbed conditions (i.e., in the absence of defective waste packages or buffer failure).

The clay baseline scenario assumes an undisturbed transport pathway, but takes only limited credit for the engineered barrier system; 95% of waste form degradation occurs in 150,000 years (i.e., fractional degradation rate of 2×10^{-5} yr^{-1}) and all waste packages fail instantaneously. Additionally, the receptor is effectively assumed to be located at the edge of the clay host rock formation, only 150 m from the repository. This differs from the receptor distance of 5,000 m used for the salt and granite baseline scenarios, so direct comparisons to those disposal options cannot be made. The resulting deterministic annual dose over 10,000,000 years is shown in Figure 4-4.

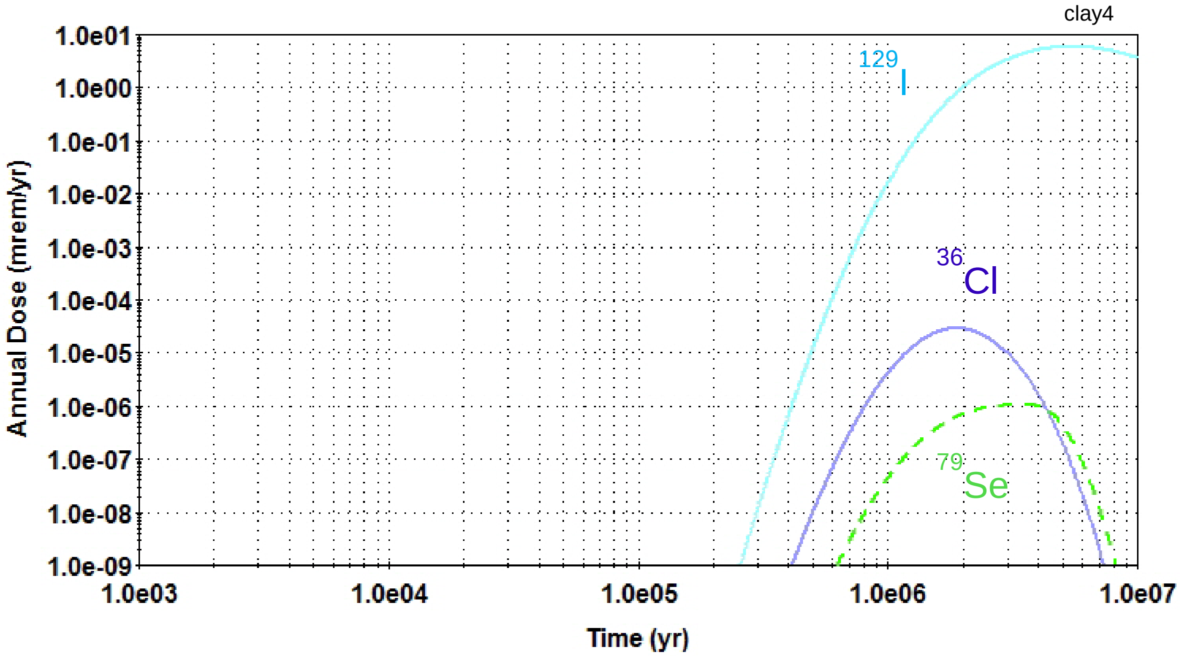


Figure 4-4. Clay Baseline Scenario Annual Dose for a Receptor 150 m from the Repository

In the first 10,000 years, the peak annual dose is 0 mrem/yr; in the first 1,000,000 years, the peak annual dose is 0.016 mrem/yr (at 1,000,000 years). The peak annual dose over the entire 10,000,000-year simulation is 5.9 mrem/yr, occurring at 5,400,000 years.

The dose is dominated by ^{129}I, with minor contributions from ^{36}Cl and ^{79}Se. As in the salt GDS model (see Section 4.4.1.1), ^{129}I and ^{36}Cl are the only two radionuclides with no sorption (k_d = 0 mL/g) throughout the disposal system, unlimited solubility, and long half-lives. ^{79}Se is a minor contributor to the dose because it is assumed to have no sorption along the 150 m flow pathway except for a small k_d of 4.6 mL/g in the 1-m thick bentonite layer. The dissolved concentration of ^{79}Se is controlled by a solubility limit, which further limits its dose contribution. Even though the initial mass of ^{79}Se (45.7 g/WP) is larger than the initial mass of ^{36}Cl (2.2 g/WP) and they have similar half-lives (290,000 yrs for ^{79}Se and 301,000 yrs for ^{36}Cl), the dose contribution from ^{36}Cl is larger due to the effects of ^{79}Se sorption in the bentonite and the ^{79}Se solubility limit.

The behavior of ^{129}I, ^{36}Cl and ^{79}Se in the deterministic safety assessment (Figure 4-4) is similar to the behavior in the FY 2011 clay GDS probabilistic model (Clayton et al. 2011, Figure 3.3-27). The deterministic annual dose supporting the safety case is generally three to four orders of magnitude higher due to a combination of significant increases from the larger inventory (~70,000 MTHM vs. 1 MTHM) and moderate decreases due to the increased clay host rock thickness (150 m vs. 65 m). The reduction in the waste package lifetime from 10,000 years to 0 years (i.e., instantaneous failure) has little effect on the dose because 10,000 years is short relative to the 350,000-year lifetime of the waste form.

In addition to ^{129}I, ^{36}Cl and ^{79}Se, several other radionuclides (e.g., ^{135}Cs, ^{237}Np, ^{242}Pu) contributed to the mean annual dose in the FY 2011 clay GDS probabilistic model (Clayton et al. 2011, Figure 3.3-27). The presence of these radionuclides as mean annual dose contributors in the FY 2011 GDS probabilistic model, but not as annual dose contributors in the deterministic safety assessment model, is due to the probabilistic treatment of the distribution coefficient, k_d, which controls the sorption of radionuclides onto the porous medium. Equation 3-1 shows the relationship between k_d, porosity, and retardation factor, R_f, where the retardation factor indicates the travel time of a sorbed radionuclide along a travel pathway relative to the travel time of a non-sorbing radionuclide (a non-sorbing radionuclide has $k_d = 0$ and $R_f = 1$). Equation 3-1 shows that the probabilistic treatment of porosity could also affect the retardation factor, for a radionuclide with a non-zero k_d.

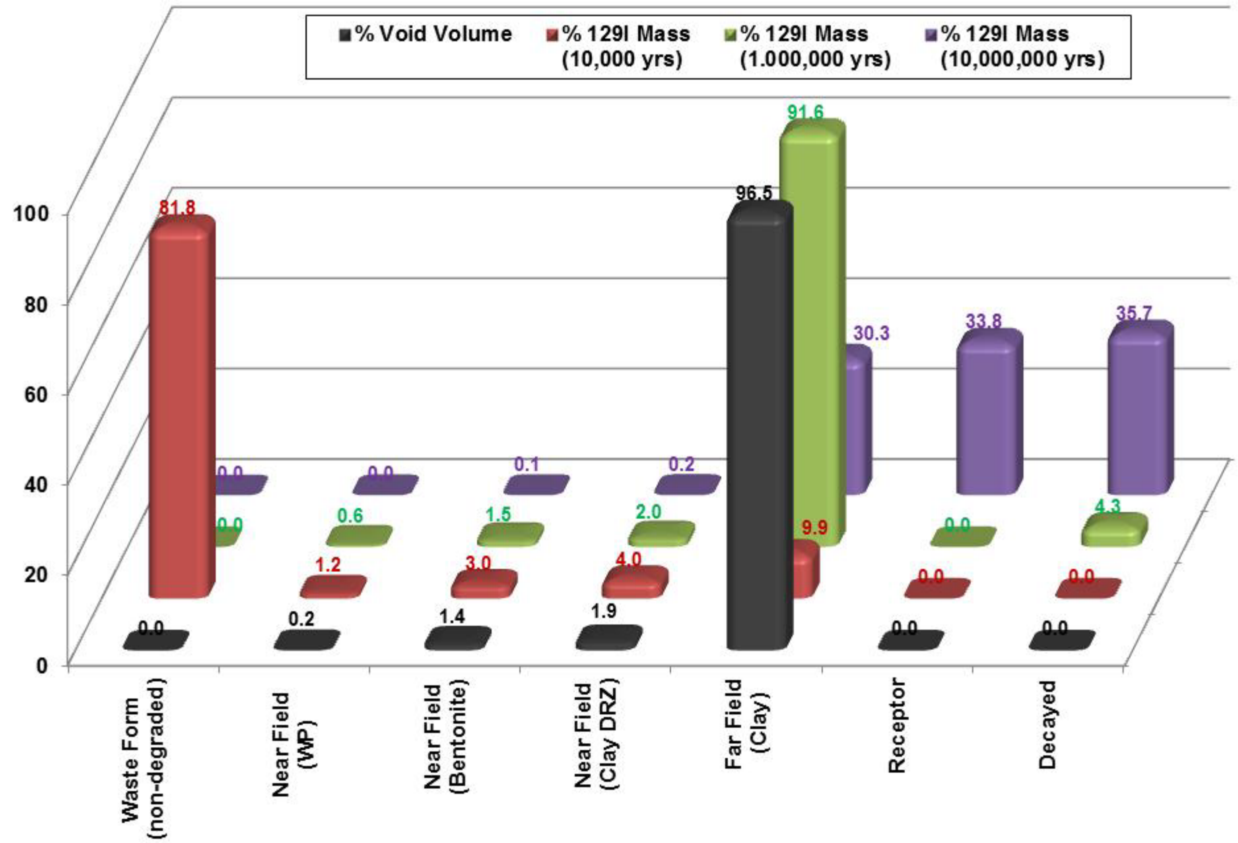
In the FY 2011 clay GDS probabilistic model, k_d values (and porosity and available porosity values) for each of 100 realizations were selected from a distribution using Monte Carlo sampling. The mean annual doses for radionuclides such as ^{135}Cs, ^{237}Np, and ^{242}Pu were dominated by realizations where low k_d values were sampled, corresponding to low retardation factors. In the far-field clay host rock, retardation factors were as low as 36 for ^{135}Cs and 83 for ^{237}Np and ^{242}Pu. These low retardation factors, combined with long half-lives, result in the minor dose contributions from ^{135}Cs, ^{237}Np, and ^{242}Pu in the probabilistic model (Clayton et al. 2011, Figure 3.3-27).

In the deterministic safety assessment model, a single k_d value was specified for each radionuclide, calculated as the mean value of the probabilistic distribution. As a result, retardation factors were 16,850 for ^{135}Cs and 37,900 for ^{237}Np and ^{242}Pu in the clay host rock. These very large retardation factors explain why ^{135}Cs, ^{237}Np, and ^{242}Pu are not dose contributors in the deterministic safety case model (Figure 4-4).

The differences in results between the FY 2011 clay GDS probabilistic model and the deterministic safety assessment model provide an indication of the sensitivity of the clay model to sorption. The sensitivity analyses in Section 4.4.2.2 further examine sorption and other processes affecting generic clay disposal system performance.

The relative distribution of ^{129}I mass in the natural and engineered barriers of the clay disposal system at three different times during the deterministic simulation supporting the safety case is shown in Figure 4-5.

At 10,000 years, 81.8% of the initial ^{129}I mass is still bound in the waste form. Less than 10% of the initial mass has been transported (by diffusion) beyond the 2.175-m thick near-field (bentonite buffer and clay DRZ) to the far-field clay host rock. At 1,000,000 years, the waste form has completely degraded and 91.6% of the initial ^{129}I mass has diffused into the 150-m thick far-field clay. The dose at 1,000,000 years (0.016 mrem/yr) is due to the small mass of ^{129}I (389 g out of an initial repository ^{129}I mass of 21,830 kg) reaching the receptor. At 10,000,000 years, 35.7% of the initial ^{129}I mass has decayed, 30.3% of the initial mass remains in the far-field clay, and 33.8% (7,370 kg) has reached the receptor location. The calculated peak annual dose (5.9 mrem/yr at 5,400,000 years) assumes, somewhat conservatively, that the entire mass from all 16,000 waste packages that is transported out of the far-field clay host rock to the overlying aquifer is captured by the pumping well at the receptor location.



NOTE: % Void Volume provides an indication of the relative distribution of water volume across the components of the disposal system. The Waste Form, Receptor, and Decay components do not have any void volume.

Figure 4-5. Distribution of ^{129}I in the Clay GDS Model Components

Based on these deterministic clay GDS model results, the following observations can be made regarding the performance of a generic clay disposal system under baseline scenario conditions:

- Radionuclide releases to the receptor location in the biosphere are small; for long-lived non-sorbing ^{129}I, releases are ~0.002% of the initial mass after 1,000,000 years and 33.8% of the initial mass after 10,000,000 years. The peak dose is 5.9 mrem/yr at 5,400,000 years.
- Radionuclide transport through the far field (clay host rock) is slow due to:
  - Diffusion-dominated transport,
  - Radionuclide sorption, and
  - Sufficient clay formation thickness (150 m).
- Radionuclide transport through the near field (the bentonite buffer and clay DRZ) is slow due to:
  - Diffusion-dominated transport,
  - Radionuclide sorption,
  - Clay DRZ healing which minimizes the potential for high-permeability fissure connections to the far-field clay.

The factors identified above as having a significant impact on near-field and far-field radionuclide transport are generally consistent with those of previous assessments for clay disposal systems, as discussed in Section 3.6 and Appendix C-3.2. For example, diffusion-dominated transport and radionuclide sorption are also cited by studies in the United States (generic), Switzerland (Opalinus Clay), France (Dossier 2005), and Belgium (SAFIR 2) as contributing to barrier performance.

Additional characteristics of a generic clay disposal system which were not captured in the deterministic clay GDS baseline scenario model include:

- Enhanced performance due to:
  - Slower waste form degradation due to reducing chemical conditions
  - Slow waste package degradation due to reducing chemical conditions
  - Increased distance to the receptor location
- Degraded performance due to:
  - Instantaneous release of gap and grain boundary inventory from the waste form
  - Increased flow rates through the near field resulting from (1) well-connected fissures, and/or (2) bentonite/clay dryout and desiccation
  - Increased flow rates through the far field

Some of these characteristics are examined in the sensitivity analyses in Section 4.4.2.2.

4.4.1.3 Deterministic Granite GDS Model Results

The deterministic granite baseline scenario is summarized in Section 4.3.2.3 and Table 4-6. Input parameters are listed in Clayton et al. (2011, Section 3.2) and Table E-3. As described in Section 4.2.3.2.3.2, under undisturbed conditions long-lived waste packages are expected to limit radionuclide releases from a granite repository.

The granite baseline scenario assumes an undisturbed transport pathway and takes only minimal credit for the waste form; 95% of waste form degradation occurs in 150,000 years (i.e., fractional degradation rate of $2{\times}10^{-5}$ yr^{-1}). The waste package provides some performance credit; only 1% (160) of the waste packages are assumed to fail instantaneously. The dose receptor is located 5,000 m from the repository. The resulting deterministic annual dose over 10,000,000 years is shown in Figure 4-6.

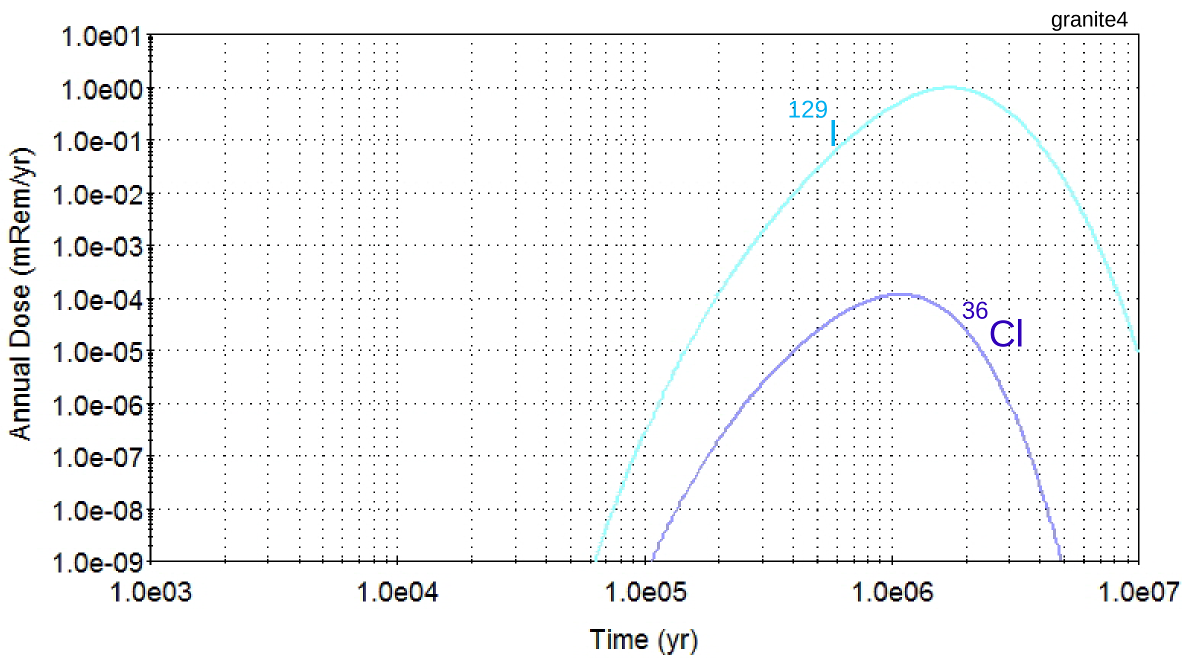


Figure 4-6. Granite Baseline Scenario Annual Dose for a Receptor 5,000 m from the Repository

In the first 10,000 years, the peak annual dose is 0 mrem/yr; in the first 1,000,000 years, the peak annual dose is 0.41 mrem/yr (at 1,000,000 years). The peak annual dose over the entire 10,000,000-year simulation is 0.95 mrem/yr, occurring at 1,730,000 years.

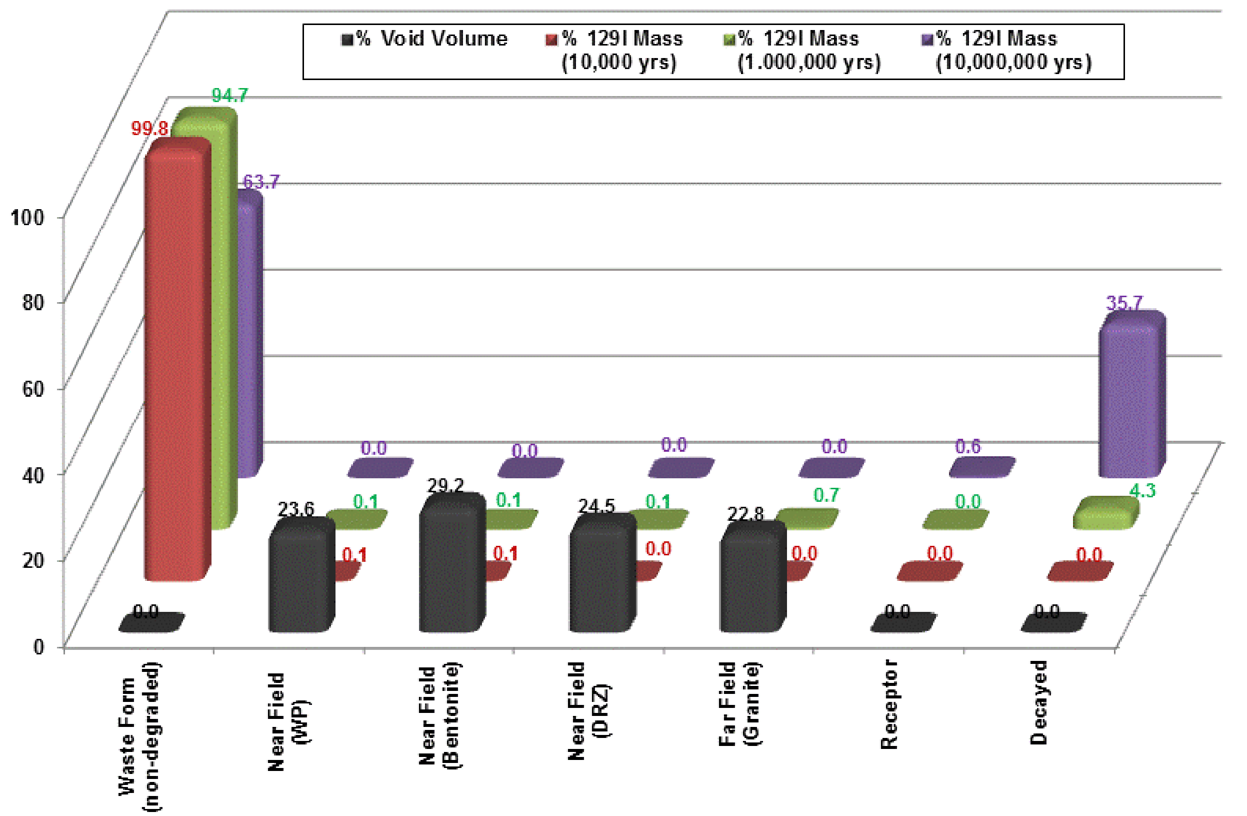
The dose is dominated by ^{129}I, with a minor contribution from ^{36}Cl. As in the salt GDS and clay GDS models (see Sections 4.4.1.1 and 4.4.1.2, respectively), ^{129}I and ^{36}Cl are the only two radionuclides with no sorption (k_d = 0 mL/g) throughout the disposal system, unlimited solubility, and long half-lives. ^{79}Se, which was a minor contributor to the clay GDS dose because it was only sorbed in the 1-m thick bentonite buffer, does not contribute to the granite GDS dose because it has a small, but non-zero, k_d of 2.0 mL/g in both the granite DRZ and the far-field granite.

The behavior of ^{129}I and ^{36}Cl in the deterministic safety assessment (Figure 4-6) is similar to the behavior in the FY 2011 granite GDS probabilistic model (Clayton et al. 2011, Figure 3.2-6). The deterministic annual dose supporting the safety case is two to three orders of magnitude higher due to (1) the faster waste form degradation rate (2×10^{-5} yr^{-1} vs. a range from 1×10^{-8} to 1×10^{-6} yr^{-1}), and (2) the reduced transverse spreading (diffusion in the bentonite buffer, mechanical dispersion in the far-field granite) due to the one-dimensional geometry used in the buffer and far field. Lesser effects result from the reduced inventory (~70,000 MTHM vs. ~90,000 MTHM), which decreases the magnitude of the dose.

In addition to ^{129}I and ^{36}Cl, several other radionuclides (e.g., ^{79}Se, ^{126}Sn) contributed to the mean annual dose in the FY 2011 granite GDS probabilistic model (Clayton et al. 2011, Figure 3.2-6). As in the clay GDS model (Section 4.4.1.2), the presence of these radionuclides as mean annual dose contributors in the FY 2011 GDS probabilistic model, but not as annual dose contributors in the deterministic safety assessment model is due to the probabilistic treatment of the distribution coefficient, k_d, which controls the sorption of radionuclides onto the porous medium. In the FY 2011 granite GDS probabilistic model, the mean annual doses for radionuclides such as^{79}Se and ^{126}Sn were dominated by realizations where low k_d values (as low as 0 for ^{126}Sn and 0.5 for ^{79}Se), corresponding to low retardation factors, were sampled in the far-field granite. These low retardation factors, combined with long half-lives, result in the minor dose contributions from ^{79}Se and ^{126}Sn in the probabilistic model (Clayton et al. 2011, Figure 3.2-6). In addition, the dissolved concentration of ^{79}Se was not controlled by a solubility limit in the FY 2011 granite GDS probabilistic model, which further enhanced its dose contribution.

An additional difference is that small doses (> 1 x 10^{-9} mrem/yr) appear as early as 2,000 years in the probabilistic model but not until about 60,000 years in the deterministic model. This difference reflects differences in the characterization of far-field flow; the early doses in the probabilistic model result from realizations where a large far-field flow velocity was sampled. The sensitivity analyses in Section 4.4.2.3 further examine sorption, flow velocity, and other processes affecting generic granite disposal system performance.

The relative distribution of ^{129}I mass in the natural and engineered barriers of the granite disposal system at three different times during the deterministic simulation supporting the safety case is shown in Figure 4-7.



NOTE: % Void Volume provides an indication of the relative distribution of water volume across the components of the disposal system. The Waste Form, Receptor, and Decay components do not have any void volume.

Figure 4-7. Distribution of ^{129}I in the Granite GDS Model Components

The effectiveness of the waste packages (1% fail instantaneously, 99% remain intact) is demonstrated by the initial ^{129}I mass that remains bound in the waste form. At 10,000 years, 99.8% of the initial ^{129}I mass is still bound in the waste form and less than 0.1% of the initial mass has been transported (by diffusion) beyond the 0.36-m thick bentonite buffer to the granite DRZ. At 1,000,000 years, 94.7% of the initial ^{129}I mass is still bound in the waste form and 0.8% of the initial mass has diffused beyond the bentonite buffer – and is mostly present in the far-field granite. The dose at 1,000,000 years (0.41 mrem/yr) is due to the small mass of ^{129}I (10.7 kg out of an initial repository ^{129}I mass of 21,830 kg) reaching the receptor. At 10,000,000 years, 35.7% of the initial ^{129}I mass has decayed, 63.7% of the initial mass remains bound in the waste form, and 0.6% (140 kg) has reached the receptor location. The calculated peak annual dose (0.95 mrem/yr at 1,730,000 years) assumes that the entire mass from all 160 failed waste packages that is transported out of the far-field granite to the overlying aquifer is captured by the drinking water well at the receptor location.

Based on these deterministic granite GDS model results, the following observations can be made regarding the performance of a generic granite disposal system under baseline scenario conditions:

- Radionuclide releases to the receptor location in the biosphere are small; for long-lived non-sorbing ^{129}I, releases are 0.05% of the initial mass after 1,000,000 years and 0.6% of the initial mass after 10,000,000 years. The peak dose is 0.95 mrem/yr at 1,730,000 years.
- Radionuclide releases from the waste form are limited by long-lived waste packages.

- Radionuclide transport through the near field (the bentonite buffer and granite DRZ) is slow due to:
  - Diffusion-dominated transport in the bentonite,
  - No defects in the buffer that produce direct connection to the far-field granite fractures, and
  - Radionuclide sorption.
- Radionuclide transport through the far field (granite fractures and matrix) is slow due to:
  - Matrix diffusion associated with fracture transport,
  - Radionuclide sorption in the matrix, and
  - Long migration distance (5,000 m) to the receptor location.

The factors identified above as having a significant impact on near-field and far-field radionuclide transport are generally consistent with those of previous assessments for granite disposal systems, as discussed in Section 3.6 and Appendix C-3.3. For example, studies in the United States (generic), Sweden (Forsmark), Finland (Posiva), and Canada (Third and Fourth Case Studies) also point to the role of waste package lifetime, diffusion, and radionuclide sorption in the engineered and natural barrier systems.

Additional characteristics of a generic granite disposal system which were not captured in the deterministic granite GDS baseline scenario model include:

- Enhanced performance due to:
  - Slower waste form degradation due to reducing chemical conditions
  - Removing the assumption that 1% of the waste packages fail instantaneously.
- Degraded performance due to:
  - Instantaneous release of gap and grain boundary inventory from the waste form
  - Increased connection to the far-field fractures due to (1) buffer erosion (failure), or (2) increased flow rates through the near field
  - Increased flow rates through the far field
  - Larger and/or better-connected far-field granite fractures
  - Reduced distance to the receptor location

Some of these characteristics are examined in the sensitivity analyses in Section 4.4.2.3.

4.4.1.4 Deterministic Deep Borehole GDS Model Results

The deterministic deep borehole baseline scenario is summarized in Section 4.3.2.4 and Table 4-7. Input parameters are listed in Clayton et al. (2011, Section 3.4) and Table E-4. As described in Section 4.2.3.2.4.2, in the absence of an advective pathway, diffusion cannot move radionuclides a significant distance through the borehole seal zone.

The deep borehole baseline scenario assumes an initial period of thermally induced advection followed by diffusion. Only minimal credit is taken for the engineered barrier system; in the disposal zone (3,000 – 5,000 m depth) 95% of waste form degradation occurs in 150,000 years (i.e., fractional degradation rate of 2×10^{-5} yr^{-1}) and all waste packages fail instantaneously, and in the seal zone (2,000 – 3,000 m depth) an annulus of disturbed rock around the borehole is assumed to enhance the effective permeability. Additionally, no credit is taken for the upper zone (0 – 2,000 m depth) of the borehole; the drinking water well that transports radionuclides to the receptor at a surface location directly above the borehole is assumed to intersect the borehole upper zone. There is no lateral distance from the borehole to the pumping well, whereas a lateral distance of 5,000 m from the repository to the pumping well is assumed in the salt and granite baseline scenarios. Therefore direct comparison to the other disposal options cannot be made.

The resulting deterministic annual dose over 10,000,000 years is shown in Figure 4-8.

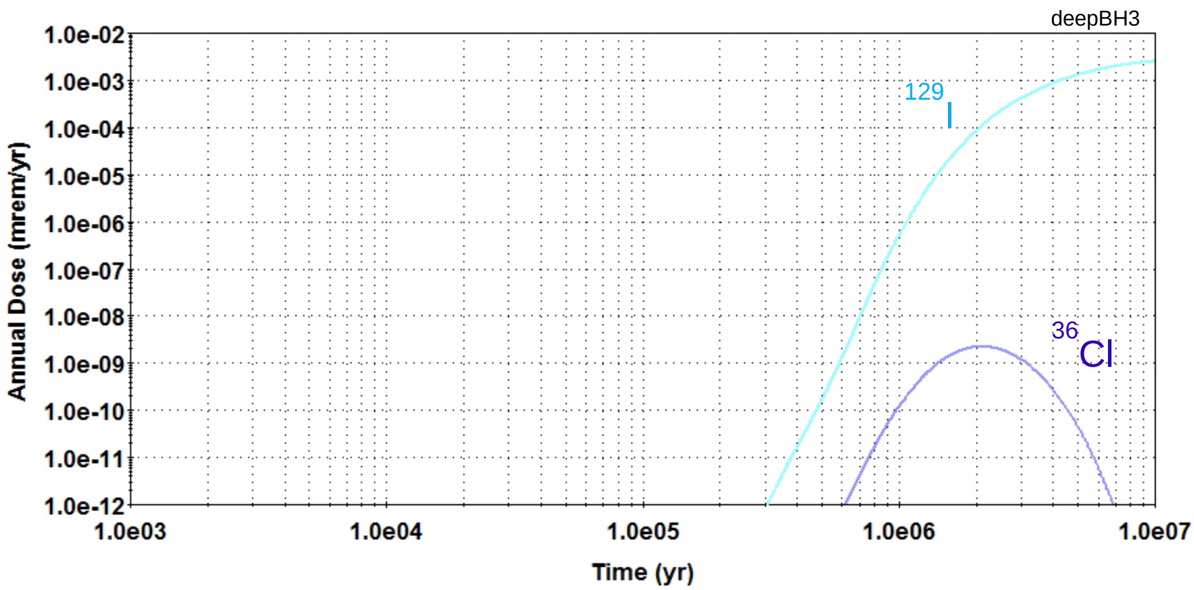


Figure 4-8. Deep Borehole Baseline Scenario Annual Dose for a Receptor Directly Above the Borehole

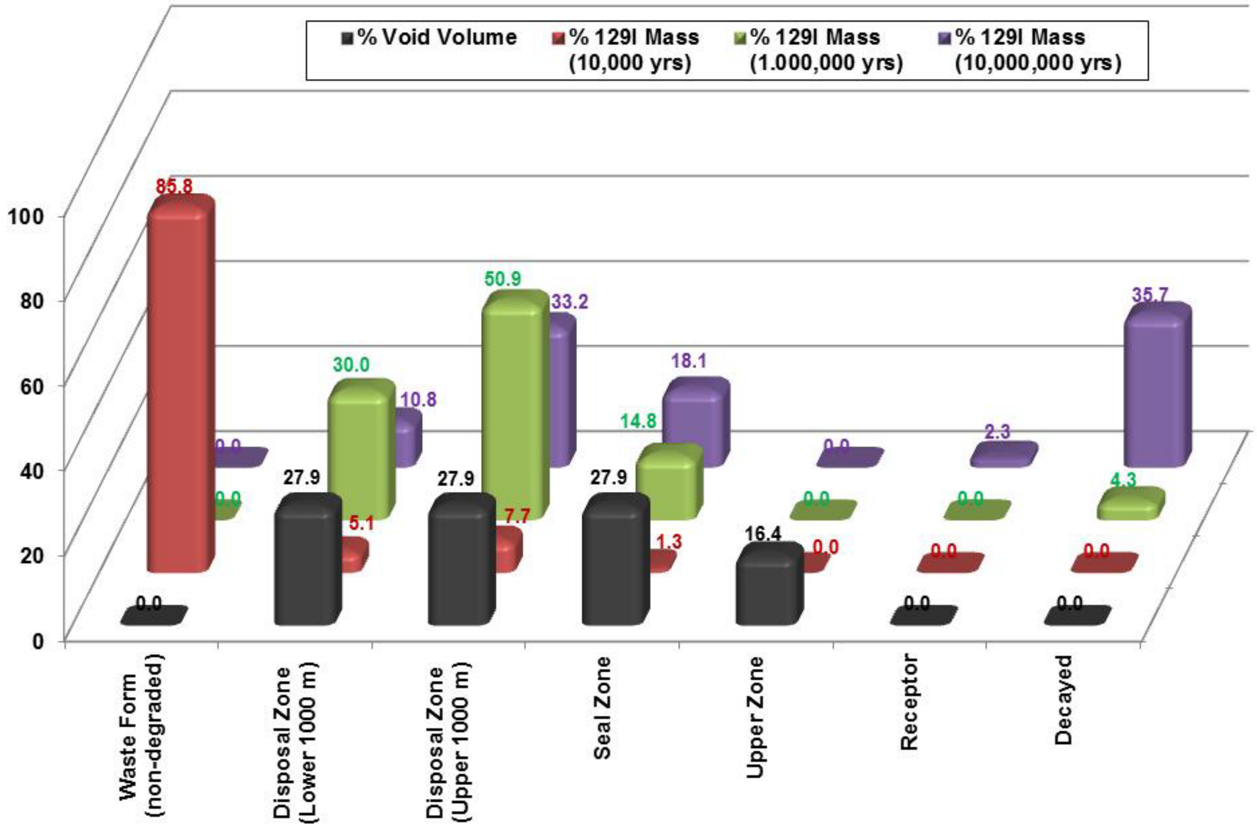
In the first 10,000 years, the peak annual dose is 0 mrem/yr; in the first 1,000,000 years, the peak annual dose is 5.1×10^{-7} mrem/yr (at 1,000,000 years). The peak annual dose over the entire 10,000,000-year simulation is 0.0025 mrem/yr, occurring at 10,000,000 years. The peak annual dose at 10,000,000 years assumes that the externally-calculated, thermally driven flow rates up the borehole at 1,000,000 years remain constant over the next 9,000,000 years.

The dose is dominated by ^{129}I, with a minor contribution from ^{36}Cl. As in the salt, clay, and granite GDS models, ^{129}I and ^{36}Cl are the only two radionuclides with no sorption (k_d = 0 mL/g) throughout the disposal system, unlimited solubility, and long half-lives. ^{79}Se, which was a minor contributor to the clay GDS dose, does not contribute to the deep borehole GDS dose because it has a small, but non-zero, k_d in all borehole zones.

The behavior of ^{129}I and ^{36}Cl in the deterministic safety assessment (Figure 4-8) is similar to the behavior in the FY 2011 deep borehole GDS probabilistic model (Clayton et al. 2011, Figure 3.4-9). The deterministic annual dose supporting the safety case is two to three orders of magnitude higher due to the faster waste form degradation rate (2×10^{-5} yr^{-1} vs. a range from 1×10^{-8} to 1×10^{-6} yr^{-1}) and an increased inventory (174 MTHM per borehole vs. 121 MTHM per borehole). The sensitivity analyses in Section 4.4.2.4 further examine processes affecting deep borehole GDS performance.

In addition to ^{129}I and ^{36}Cl, ^{99}Tc also contributed to the mean annual dose in the FY 2011 deep borehole GDS probabilistic model (Clayton et al. 2011, Figure 3.4-9). As in the clay and granite GDS models, the additional dose contribution in the FY 2011 GDS probabilistic model is due to the probabilistic treatment of the distribution coefficient, k_d, which controls the sorption of radionuclides onto the porous medium. In the FY 2011 deep borehole GDS probabilistic model, the mean annual dose for ^{99}Tc was dominated by realizations where low k_d values were sampled (as low as 0.00001 mL/g in the disposal zone and 0.0001 in the seal zone), corresponding to low retardation factors. In the deterministic safety case model, the k_d values were 1.7 mL/g in the disposal zone and 17 in the seal zone.

The relative distribution of ^{129}I mass in the natural and engineered barriers of the deep borehole disposal system at three different times during the deterministic simulation supporting the safety case is shown in Figure 4-9.



NOTE: % Void Volume provides an indication of the relative distribution of water volume across the components of the disposal system. The Waste Form, Receptor, and Decay components do not have any void volume.

Figure 4-9. Distribution of ^{129}I in the Deep Borehole GDS Model Components

At 10,000 years, 85.8% of the initial ^{129}I mass is still bound in the waste form. Only 1.3% of the initial mass has been transported (by thermally induced advection) out of the disposal zone and into the 1,000-m thick seal zone. At 1,000,000 years, the waste form has completely degraded, and 14.8% of the initial ^{129}I mass has been transported into the seal zone. The transport of ^{129}I out of the disposal zone is advection dominated in the first approximately 300,000 years until the thermally induced flow rates have declined significantly. After about 700,000 years, the transport of ^{129}I out of the disposal zone is diffusion dominated. Further away from the thermal effects of the disposal zone, transport is diffusion dominated even in the first 300,000 years. The small dose at 1,000,000 years (5.1×10^{-7} mrem/yr) is due to a small mass of ^{129}I (5.1×10^{-3} g out of an initial borehole disposal zone ^{129}I mass of 54.6 kg) actually reaching the receptor. At 10,000,000 years, 35.7% of the initial ^{129}I mass has decayed, 44.0% of the initial mass remains in the disposal zone, 18.1% is in the seal zone, and 2.3% (1.3 kg) has reached the receptor location. The calculated peak annual dose (0.0025 mrem/yr at 10,000,000 years) assumes that all non-sorbing radionuclides leaving the seal zone are rapidly transported through the upper zone to the receptor.

Based on these deterministic deep borehole GDS model results, the following observations can be made regarding the performance of a generic deep borehole disposal system under baseline scenario conditions:

- Radionuclide releases to the receptor location in the biosphere (directly above the borehole at the surface) are small; for long-lived non-sorbing ^{129}I, releases are ~0.000001% of the initial mass after 1,000,000 years and 2.3% of the initial mass after 10,000,000 years. The peak dose is 0.0025 mrem/yr at 10,000,000 years.

- Radionuclide transport through the bentonite seal zone is slow due to:
  - Very low thermally induced fluid flow rates resulting in diffusion-dominated transport,
  - Durability of the seals with only minor DRZ bypass,
  - Radionuclide sorption, and
  - Long migration distance (1,000 m).
- Radionuclide transport through the disposal zone is slow due to:
  - Low thermally induced fluid flow rates that decrease over time, resulting in diffusion-dominated transport after about 700,000 years,
  - Radionuclide sorption, and
  - Long migration distance (as much as 2,000 m) for the deepest waste packages.
- Radionuclide transport through the basement deep granite is negligible due to:
  - Very low permeability and lack of significant fracture connection to overlying formations.

The factors identified above as having a significant impact on radionuclide transport up the borehole are generally consistent with those of the generic study for deep borehole disposal systems, as discussed in Section 3.6 and Appendix C-3.4. Conducted in the United States, this study also cites diffusion-dominated transport, radionuclide sorption, and the long migration distance as contributing to barrier performance.

Additional characteristics of a generic deep borehole disposal system which were not captured in the deterministic deep borehole GDS baseline scenario model include:

- Enhanced performance due to:
  - Slower waste form degradation due to reducing chemical conditions
  - Removing the model assumption that all radionuclides diffuse upward through the borehole; the dose would be lower if some radionuclides diffused radially outward into deep granite host rock that was isolated from shallower formations.
  - Continually declining thermally driven flow rates up the borehole between 1,000,000 and 10,000,000 years
  - Radionuclide delay in the borehole upper zone
- Degraded performance due to:
  - Instantaneous release of gap and grain boundary inventory from the waste form
  - Increased transport through the seal zone (1) a more significant DRZ bypass, and/or (2) more significant bentonite seal degradation

Some of these characteristics are examined in the sensitivity analyses in Section 4.4.2.4.

4.4.2 Baseline Scenario Sensitivity Analyses

As discussed in Section 2.3.1, estimates of disposal system performance must consider uncertainty. As a geologic disposal system program matures, uncertainties in disposal system performance models are typically treated using probabilistic methods. However, to support this initial safety case, the effects of uncertainties in the safety assessments for the four baseline scenarios are investigated using sensitivity analyses in the form of one-off deterministic simulations. A one-off simulation is performed by changing the value of a single uncertain parameter from its baseline value to other values within its distribution, or to a reasonable bounding value, and examining the corresponding effect on system performance. These one-off sensitivity simulations, described in the following subsections, provide additional insights into which parameters, features, and/or barriers contribute to the overall capability of a specific disposal system to isolate waste from the biosphere under the assumed baseline scenario conditions. Sensitivities are examined with respect to impact on ^{129}I release and migration because ^{129}I is the dominant (and in some cases the only) contributor to annual dose. The insights provided by these deterministic one-off simulations are informative during generic safety assessment analyses. As safety assessments become more site specific and more information describing parameter values becomes available, probabilistic sensitivity analyses provide more detailed insights, particularly into the effects of couplings between uncertain processes and parameters.

4.4.2.1 Salt GDS Model Sensitivity Analyses

Annual dose results from the deterministic salt GDS baseline scenario are shown in Figure 4-2. The following one-off sensitivity simulations were performed to investigate the effects on ^{129}I movement through the disposal system:

- Waste form fractional degradation rate (Figure 4-10)
- Integrity of the near-field salt DRZ between the repository and the underlying interbed (Figure 4-11)
- Brine flow rate in the engineered barrier system, DRZ, and anhydrite interbed (Figure 4-12)
- Molecular diffusion coefficient (Figure 4-13)
- Sorption (^{129}I distribution coefficient) in the anhydrite interbed (Figure 4-14)
- Distance to receptor location (Figure 4-15)

Waste Form Degradation—The effect of waste form degradation rate on ^{129}I annual dose is shown in Figure 4-10. The sensitivity analysis includes three fractional degradation rate cases:

- Fast Waste Form Degradation (0.1 yr^{-1})—100% of the radionuclide mass is released from the waste form in the first 250 years. This provides a bounding case for instantaneous release of gap and grain boundary inventory from the waste form. An estimate of the ^{129}I gap fraction from used fuel is the following: 0.0204 (minimum); 0.1124 (most likely); 0.2675 (maximum) (Sandia National Laboratories 2008c, Table 6.3.7-29).
- Baseline Waste Form Degradation (2×10^{-5} yr^{-1})—50% of the radionuclide mass is released from the waste form in the first 35,000 years, 95% of the mass is released by 150,000 years, and 99.9% of the mass is released by about 350,000 years.
- Slow Waste Form Degradation (1×10^{-7} yr^{-1})—50% of the radionuclide mass is released from the waste form after 4,800,000 years, and 76% of the mass is released by 10,000,000 years. This slow degradation rate, consistent with reducing chemical conditions, was the assumed to be the most likely rate in the FY 2011 salt, granite, and deep borehole GDS models (Clayton et al. 2011).

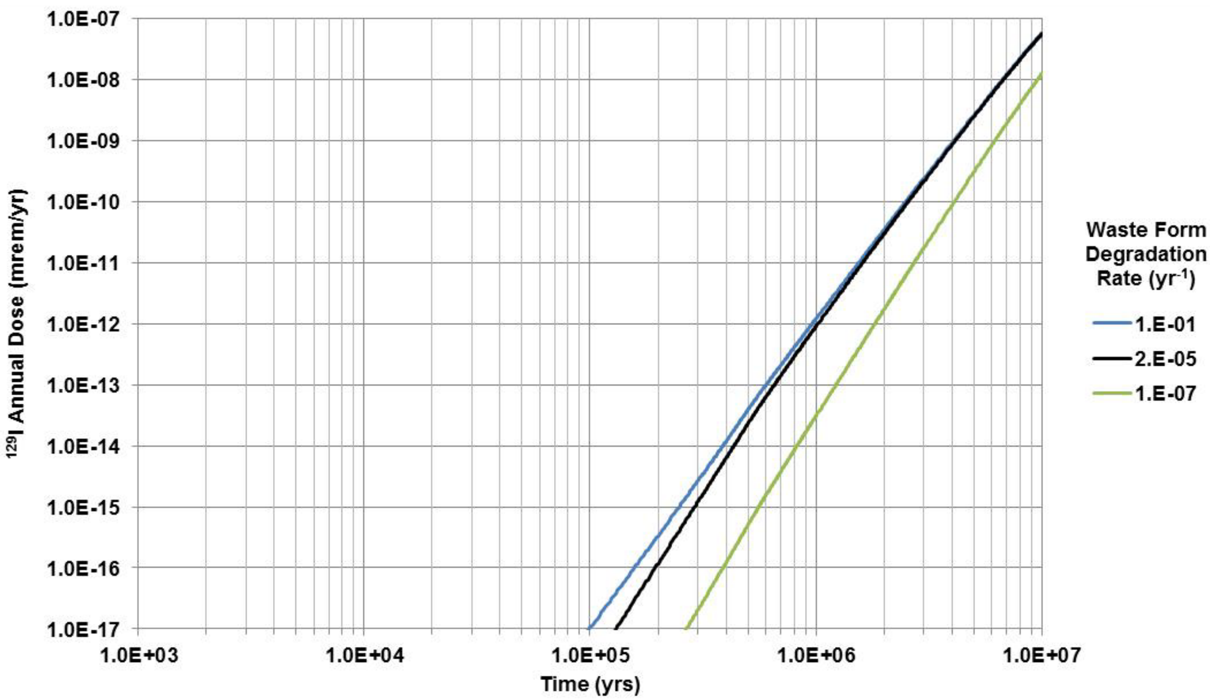


Figure 4-10. Effect of Waste Form Degradation Rate on Annual Dose from ^{129}I in the Salt GDS Model

In the fast degradation rate (0.1 yr^{-1}) case, all mass is released (degraded) from the waste form in the first 250 years. This is roughly equivalent to assuming that all ^{129}I mass is instantaneously released as gap and grain boundary inventory (i.e., a ^{129}I gap fraction of 1.0 as compared to the estimated range of 0.0204–0.2675). The mass released from the waste form diffuses vertically downward through the repository and 5-m thick near-field salt DRZ, and into the 1-m thick interbed underlying the repository. This vertical diffusion occurs across the repository footprint (a 2,146-m per side square with a porosity of 0.039) corresponding to a diffusion area of approximately 180,000 m^2. In the fast degradation rate case, 73% of the initial ^{129}I mass reaches the near-field salt DRZ and underlying interbed by 100,000 years, whereas in the baseline case, only 57% of the initial mass reaches the DRZ and interbed by 100,000 years, and 22% of the mass is still undegraded. Despite the greater early transport of ^{129}I mass away from the repository in the fast degradation rate case, the effect on annual dose is not significant. This is because, in a diffusion-dominated system, the effect of the early mass on annual dose is attenuated in the 5,000-m far-field interbed by (1) diffusive fluxes into the interbed that decline over time as a function of the concentration gradient, (2) the conceptual model assumptions about the system geometry, and (3) slow diffusive travel times through the interbed.

In the fast degradation rate case, the early mass results in a larger diffusive flux into the interbed at early time. However, this concentration gradient quickly equilibrates. Conversely, in the baseline case, the early-time diffusive flux is not as large, but the concentration equilibration is slower. Over longer timescales, the cumulative flux into the interbed is similar in the two cases. The effect of the larger early-time diffusive flux in the fast degradation case is further limited by the system geometry. While the vertical diffusion from the DRZ to the interbed occurs across a 180,000-m^2 diffusion area corresponding the repository footprint void volume, the subsequent horizontal diffusion along the interbed toward the receptor location is across a 21.5-m^2 diffusion area corresponding to the interbed cross-section void volume (2,146-m wide by 1-m thick with a porosity of 0.01). Therefore, the large early-time concentration gradient due to the extra early mass present in the underlying interbed in the fast degradation rate case has a limited effect on the diffusion rate along the interbed (ranging from 0.001 to 0.01 g/yr) due to the relatively small diffusion area, resulting in only about an extra 0.4 kg of ^{129}I (out of an initial mass of 21,830 kg) in the interbed after 100,000 years and an extra 0.2 kg after 10,000,000 years. And finally, in both the baseline and fast waste form degradation cases, the degradation time is shorter than the travel time horizontally along the 5,000-m interbed; therefore, the horizontal travel time through the interbed is the dominant process, and increases in the waste form degradation rate have little effect on annual dose.

For the slow fractional degradation rate (1×10^{-7} yr^{-1}), 50% of the radionuclide mass is not released from the waste form until 4,800,000 years, and only 76% of the mass is released by 10,000,000 years. In this case, the degradation time is slower than the travel time through the far-field interbed and the effect of the slower degradation rate is to reduce the peak dose by about a factor of 4 at 10,000,000 years.

Near-Field DRZ Integrity—The effect of the integrity of the near-field salt DRZ on ^{129}I annual dose is shown in Figure 4-11. The sensitivity analysis includes two cases:

- <u>Baseline Intact Near-Field Salt</u>**—**The near-field salt DRZ between the repository and the underlying interbed is 5-m thick and lacks any fast fracture pathways. The brine flow rate is low enough that transport through the DRZ is diffusion dominated. Specific flow values are described below in the discussion of sensitivity to flow rate.
- <u>Damaged Near-Field Salt</u>**—**The effective thickness of the near-field salt DRZ is 1 m and a multiplier of 1,000 is applied to the baseline brine flow rate history. The brine flow multiplier results in advective transport through the DRZ. These enhanced transport properties are considered to represent the effects of better-connected, non-healing fractures between the repository and the underlying interbed.

The effect of the more damaged near-field salt DRZ has a minor effect on annual dose, increasing the peak dose by about a factor of 3 at 10,000,000 years.

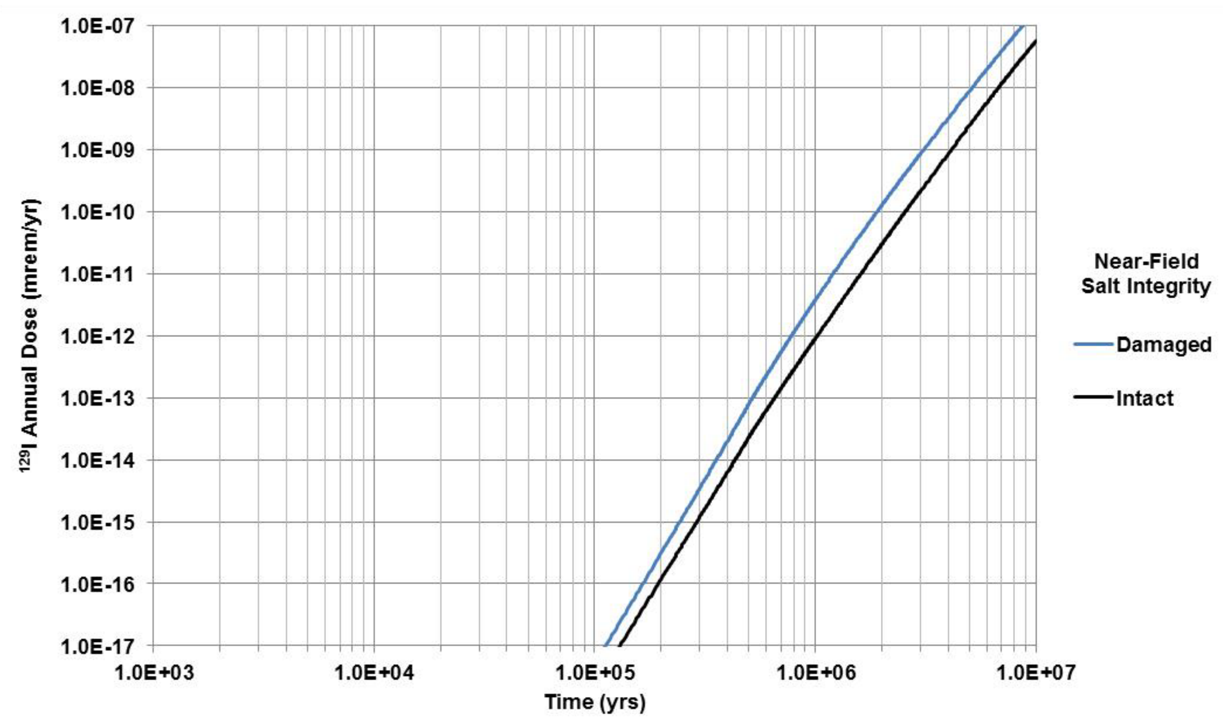


Figure 4-11. Effect of Near-Field DRZ Integrity on Annual Dose from ^{129}I in the Salt GDS Model

Brine Flow Rate—The effect of the brine flow rate on ^{129}I annual dose is shown in Figure 4-12. The sensitivity analysis includes three brine flow cases:

- Baseline Brine Flow—The brine flow rate through the engineered barrier system, near-field salt DRZ, and interbed is based on a single flow rate history realization from Clayton et al. (2011, Section 3.1.3). The baseline flow history, summarized below in Table 4-8, results in diffusion-dominated transport throughout the disposal system.
- Brine Flow Increased by a Factor of 10—A multiplier of 10 is applied to the baseline brine flow rate histories in all regions. The increased brine flow represents the potential effects of repository pressurization from creep closure and gas generation and/or higher permeability. These increased flow rates result in advective transport that is of the same order of magnitude as diffusive transport.
- Brine Flow Increased by a Factor of 100—A multiplier of 100 is applied to the baseline brine flow rate histories in all regions. These increased flow rates result in advection-dominated transport throughout the system.

The case where the brine flow rates are multiplied by 100 has a much more significant effect on dose than when the flow rates are multiplied by 10. This is because the factor-of-100 multiplier changes the disposal system to advection-dominated transport. A significant flow rate increase at around 500,000 years drives the increase in annual dose for the case with the factor-of-100 multiplier. The effects of the factor-of-10 brine flow multiplier are much smaller (only about a factor of 2 increase in dose) because the increase in flow is only increasing the contribution from advective transport to a level similar to that already provided by diffusive transport.

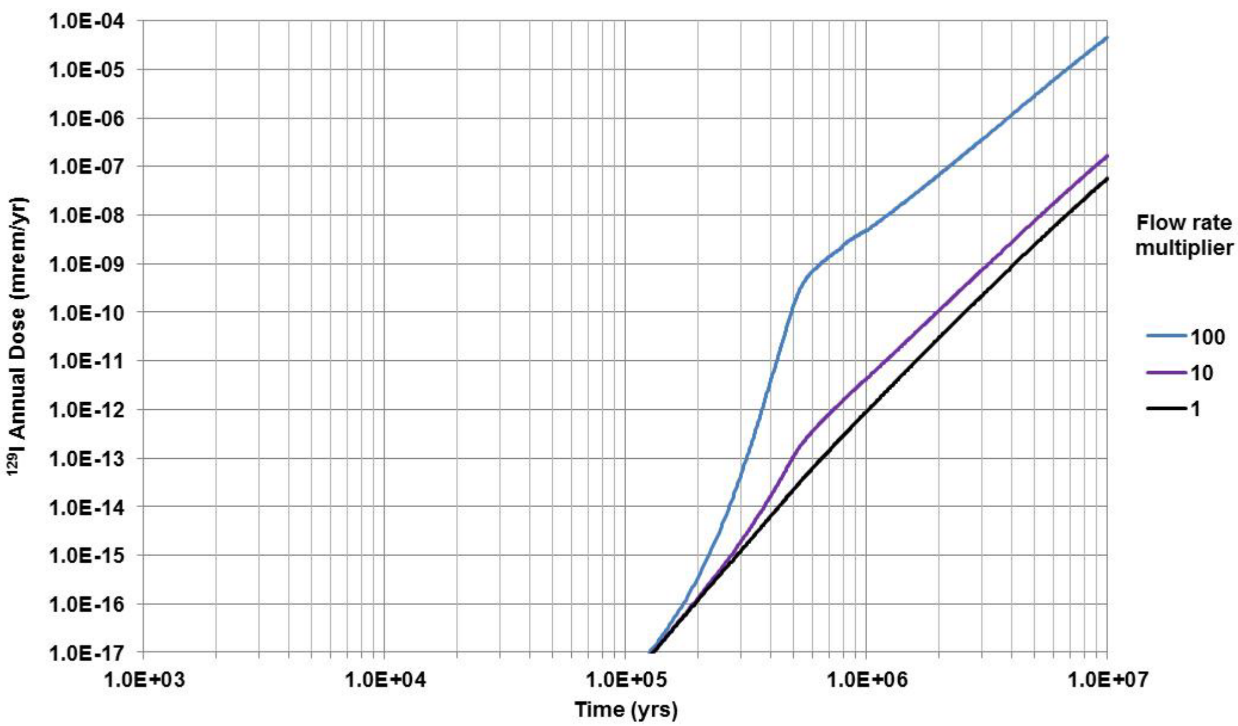


Figure 4-12. Effect of Brine Flow Rate on Annual Dose from ^{129}I in the Salt GDS Model

Table 4-8. Summary of the Baseline Brine Flow

Time (yrs)	Darcy Velocity in EBS and Near-Field DRZ (m/yr)	Darcy Velocity in Interbed (m/yr)
0	0	0
9,000	0	0
10,000	3.36×10^{-9}	2.79×10^{-14}
100,000	2.57×10^{-6}	6.92×10^{-9}
480,000	8.89×10^{-6}	1.47×10^{-6}
1,000,000	8.56×10^{-7}	3.96×10^{-7}
10,000,000	8.56×10^{-7}	3.96×10^{-7}

Diffusion—The effect of the diffusion coefficient on ^{129}I annual dose is shown in Figure 4-13. The sensitivity analysis includes two cases:

- Baseline Molecular (Free Water) Diffusion Coefficient (2.30×10^{-9} m^2/s)—The corresponding effective diffusion coefficient for ^{129}I in each region is based on the local porosity and tortuosity.
- Enhanced Molecular (Free Water) Diffusion Coefficient (1.15×10^{-8} m^2/s)—This results in a corresponding effective diffusion coefficient for ^{129}I in each region that is a factor of 5 larger than the baseline value. The increased molecular diffusion coefficient reproduces potential changes in the effective diffusion coefficient for ^{129}I that might result from changes in available porosity (e.g., due to anion exclusion) or tortuosity.

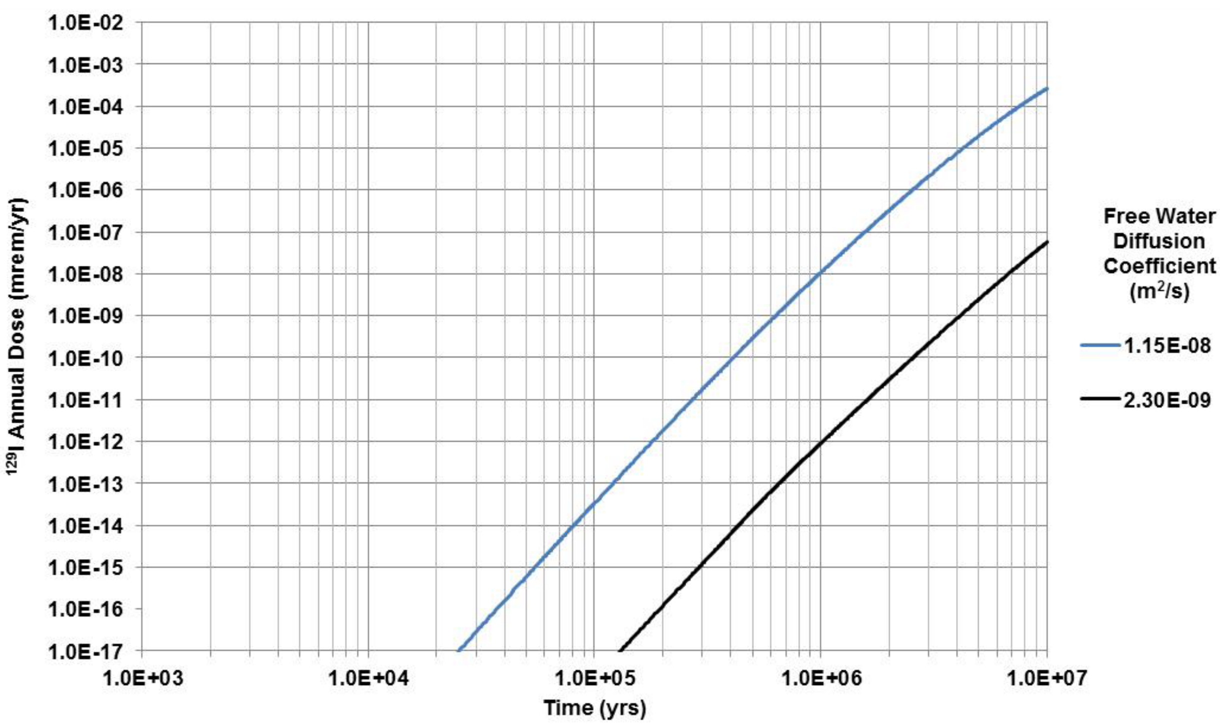


Figure 4-13. Effect of Diffusion Coefficient on Annual Dose from ^{129}I in the Salt GDS Model

The factor-of-5 increase in diffusion coefficient has a significant effect on dose. This is because of the corresponding factor-of-5 increase in diffusive flux rate in a diffusion-dominated system, which shifts the dose curve to the left by a factor of five on the time axis.

Interbed Sorption—The effect of sorption in the interbed on ^{129}I annual dose is shown in Figure 4-14. The sensitivity analysis includes three cases:

- Baseline ^{129}I Sorption (k_d = 0.00 mL/g)—The corresponding retardation factor in the interbed is 1.0.
- Increased ^{129}I Sorption (k_d = 0.01 mL/g)—The corresponding retardation factor in the interbed is 3.5.
- Increased ^{129}I Sorption (k_d = 0.10 mL/g)—The corresponding retardation factor in the interbed is 26.0.

Changes in ^{129}I k_d have a significant effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. For the case with k_d = 0.01 mL/g and R_f = 3.5, the dose curve shifts to the right by a factor of 3.5 on the time axis. For the case with k_d = 0.10 mL/g and R_f = 26.0, the dose curve shifts to the right by a factor of 26 on the time axis.

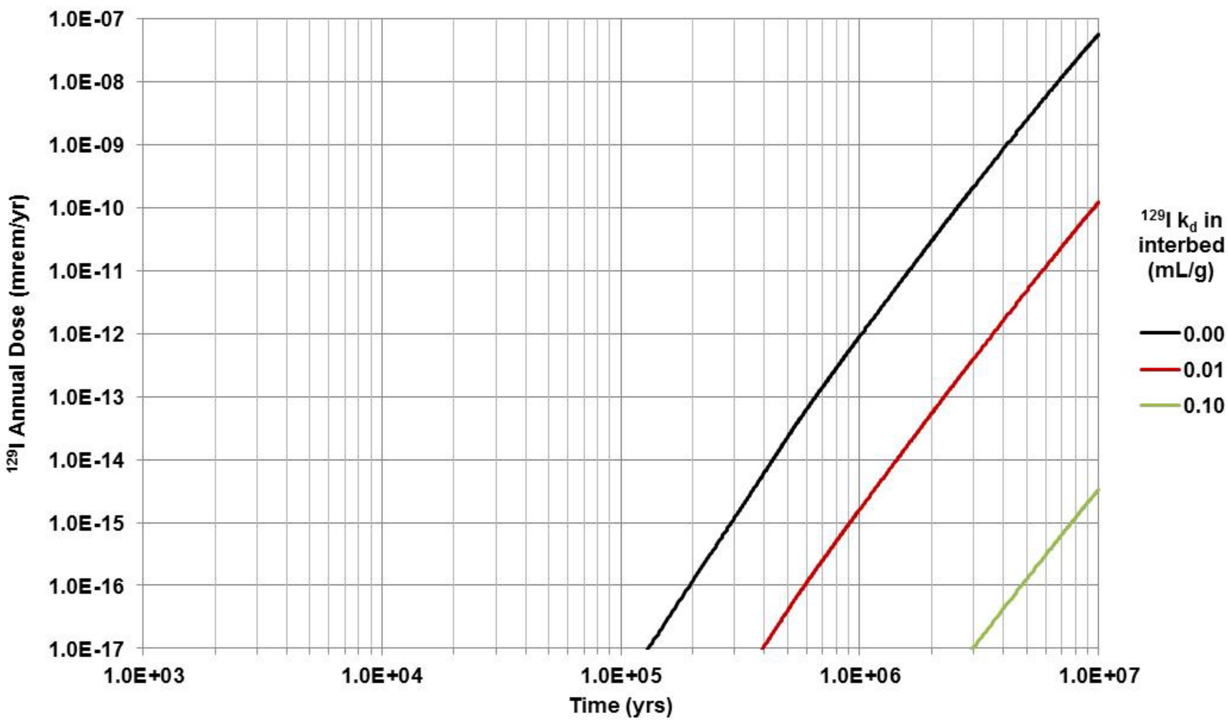


Figure 4-14. Effect of Interbed Sorption on Annual Dose from ^{129}I in the Salt GDS Model

Receptor Distance—The effect of distance to the receptor on ^{129}I annual dose is shown in Figure 4-15. The sensitivity analysis includes three cases:

- Baseline interbed length to receptor (5,000 m)
- Reduced interbed length to receptor (3,000 m)
- Reduced interbed length to receptor (1,000 m)

The annual dose is quite sensitive to the distance to the receptor. The effects of reducing the interbed length are greater than linear because, in these salt disposal system simulations, diffusion is the dominant transport mechanism in the interbed and the peak dose is controlled by the leading edge of the diffusion front. The sensitivity case with a 1,000-m interbed length also provides an indication of the potential dose from a vertical diffusion pathway of 1,000 m (i.e., corresponding to a 1,000-m deep repository).

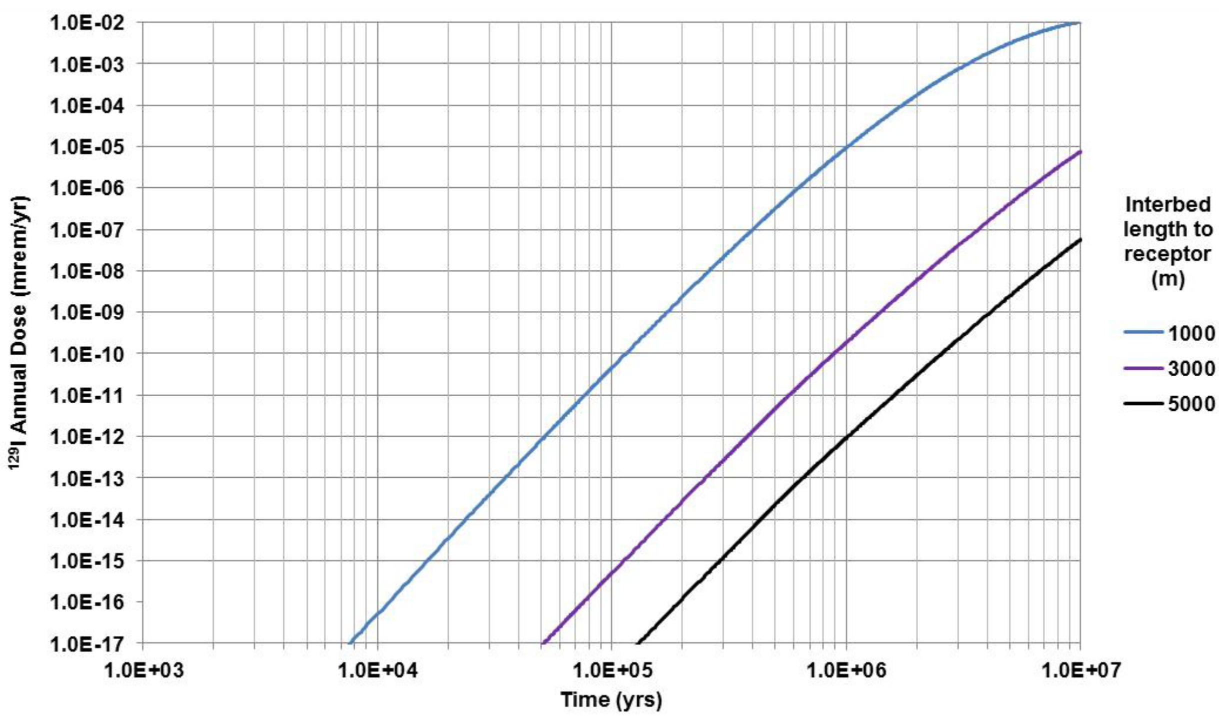


Figure 4-15. Effect of Distance to Receptor on Annual Dose from ^{129}I in the Salt GDS Model

Summary—Based on these six sensitivity analyses examining "one-off" conditions from the baseline scenario, the following observations can be made regarding the performance of a generic salt disposal system under baseline scenario conditions:

- Processes and parameters affecting radionuclide transport through the 5,000-m far-field interbed can have a significant effect on annual dose. These include sorption, k_d, and distance to receptor.
- Processes and parameters affecting radionuclide transport through the entire salt disposal system can have a significant effect on annual dose. These include brine flow rate and diffusion coefficient. These system-wide effects are most important in the far-field interbed.
- Processes and parameters affecting waste form degradation can have a moderate effect on annual dose. Increasing the degradation rate does not significantly increase the dose because the effects are mitigated by slow diffusion into and through the far-field interbed. Decreasing the degradation rate decreases the annual dose.
- Processes and parameters affecting radionuclide transport through the 5-m near-field salt DRZ have a minimal effect on dose.

4.4.2.2 Clay GDS Model Sensitivity Analyses

Annual dose results from the deterministic clay GDS baseline scenario are shown in Figure 4-4. The following one-off sensitivity simulations were performed to investigate the effects on ^{129}I movement through the disposal system:

- Waste form fractional degradation rate (Figure 4-16)
- Waste package lifetime (Figure 4-17)
- Integrity of the bentonite buffer and DRZ clay (Figure 4-18)
- Flow rate in the engineered barrier system and far field (Figure 4-19)
- Molecular diffusion coefficient (Figure 4-20)
- Sorption (^{129}I distribution coefficient) in the far-field clay (Figure 4-21)
- Thickness of the far-field clay (Figure 4-22)

Waste Form Degradation—The effect of waste form degradation rate on ^{129}I annual dose is shown in Figure 4-16. The sensitivity analysis includes the same three fractional degradation rate cases described in Section 4.4.2.1 for the salt analyses:

- <u>Fast Waste Form Degradation (0.1 yr^{-1})</u>—100% of the radionuclide mass is released from the waste form in the first 250 years. This provides a bounding case for instantaneous release of gap and grain boundary inventory from the waste form.
- <u>Baseline Waste Form Degradation (2×10^{-5} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form in the first 35,000 years, 95% of the mass is released by 150,000 years, and 99.9% of the mass is released by about 350,000 years.
- <u>Slow Waste Form Degradation (1×10^{-7} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form after 4,800,000 years, and 76% of the mass is released by 10,000,000 years.

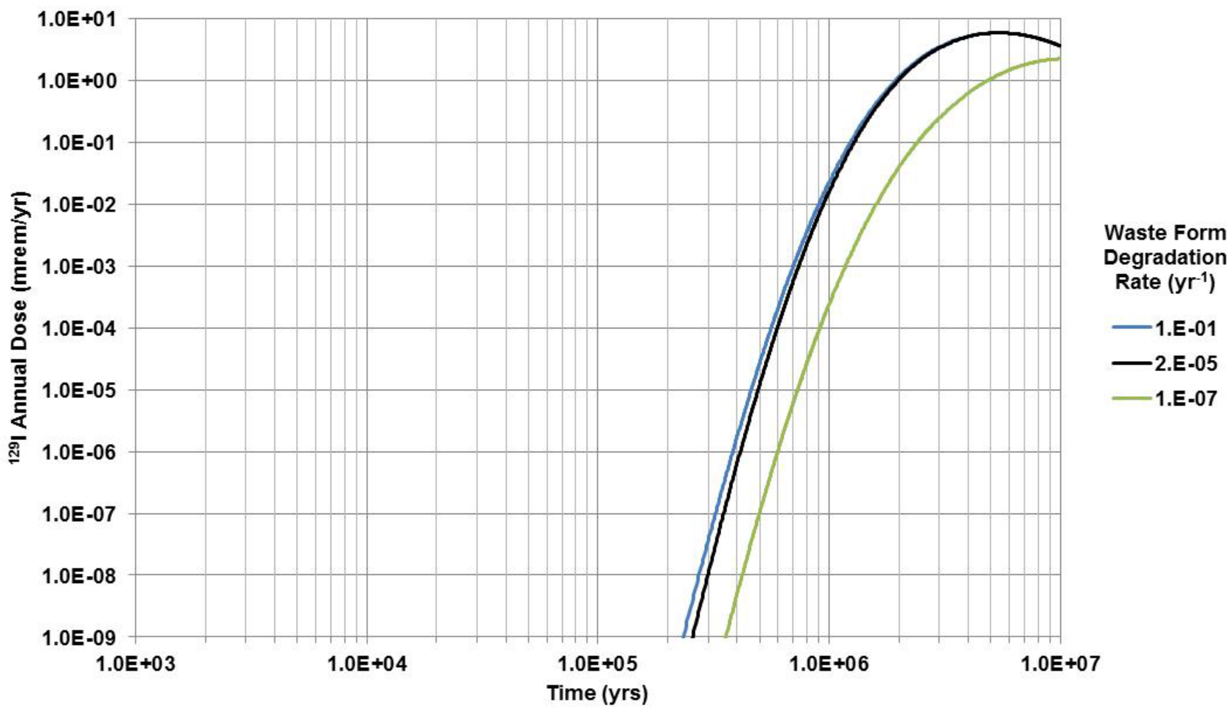


Figure 4-16. Effect of Waste Form Degradation Rate on Annual Dose from ^{129}I in the Clay GDS Model

In the fast degradation rate (0.1 yr^{-1}) case, all mass is released (degraded) from the waste form in the first 250 years. This is roughly equivalent to assuming that all ^{129}I mass is instantaneously released as gap and grain boundary inventory (i.e., a ^{129}I gap fraction of 1.0 as compared to the estimated range of 0.0204–0.2675). The mass released from the waste form diffuses vertically through the 1.025-m thick bentonite buffer and the 1.15-m thick fissured clay DRZ, and into the 150-m thick clay host rock. In the fast degradation rate case, 86% of the initial ^{129}I mass reaches the far-field clay host rock by 100,000 years, whereas in the baseline case, only 71% of the initial mass reaches the far-field clay by 100,000 years, and 14% of the mass is still undegraded. Despite the greater early transport of ^{129}I mass away from the repository in the fast degradation rate case, the effect on annual dose is not significant. This is because, in a diffusion-dominated system, the effect of the early mass on annual dose is attenuated in the 150-m far-field clay by (1) diffusive fluxes into the far field that decline over time as a function of the concentration gradient, and (2) slow diffusive travel times through the far-field clay.

In the fast degradation rate case, the early mass results in a larger diffusive flux into the far-field clay at early time. However, this concentration gradient quickly equilibrates. Conversely, in the baseline case, the early-time diffusive flux is not as large, but the concentration equilibration is slower. Over longer timescales, the cumulative flux into the far field is similar in the two cases. Also, in both the baseline and fast waste form degradation cases, the degradation time is shorter than the travel time through the 150-m far-field clay; therefore, the travel time through the far-field clay is the dominant process, and increases in the waste form degradation rate have little effect on annual dose.

For the slow fractional degradation rate (1×10^{-7} yr^{-1}), 50% of the radionuclide mass is not released from the waste form until 4,800,000 years, and only 76% of the mass is released by 10,000,000 years. In this case, the degradation time is slower than the travel time through the far-field clay and the effect of the slower degradation rate is to reduce the magnitude of the peak dose by about a factor of 3 and delay the time of the peak dose by about a factor of 2.

Waste Package Degradation—The effect of waste package lifetime on ^{129}I annual dose is shown in Figure 4-17. The sensitivity analysis includes four cases:

- Baseline Waste Package Lifetime (0 years)—All 16,000 waste packages fail instantaneously, no performance credit for the waste package.
- Moderate Waste Package Lifetime (100,000 years)—All 16,000 waste packages fail at 100,000 years.
- Long Waste Package Lifetime (500,000 years)—All 16,000 waste packages fail at 500,000 years.
- Very Long Waste Package Lifetime (1,000,000 years)—All 16,000 waste packages fail at 1,000,000 years.

The effect of waste package lifetime on system performance is to delay the onset of waste form degradation and radionuclide release from the waste form. The delay is evident in the annual dose curves; they are all shifted to the right on the time axis (100,000, 500,000, and 1,000,000 years) relative to the baseline case.

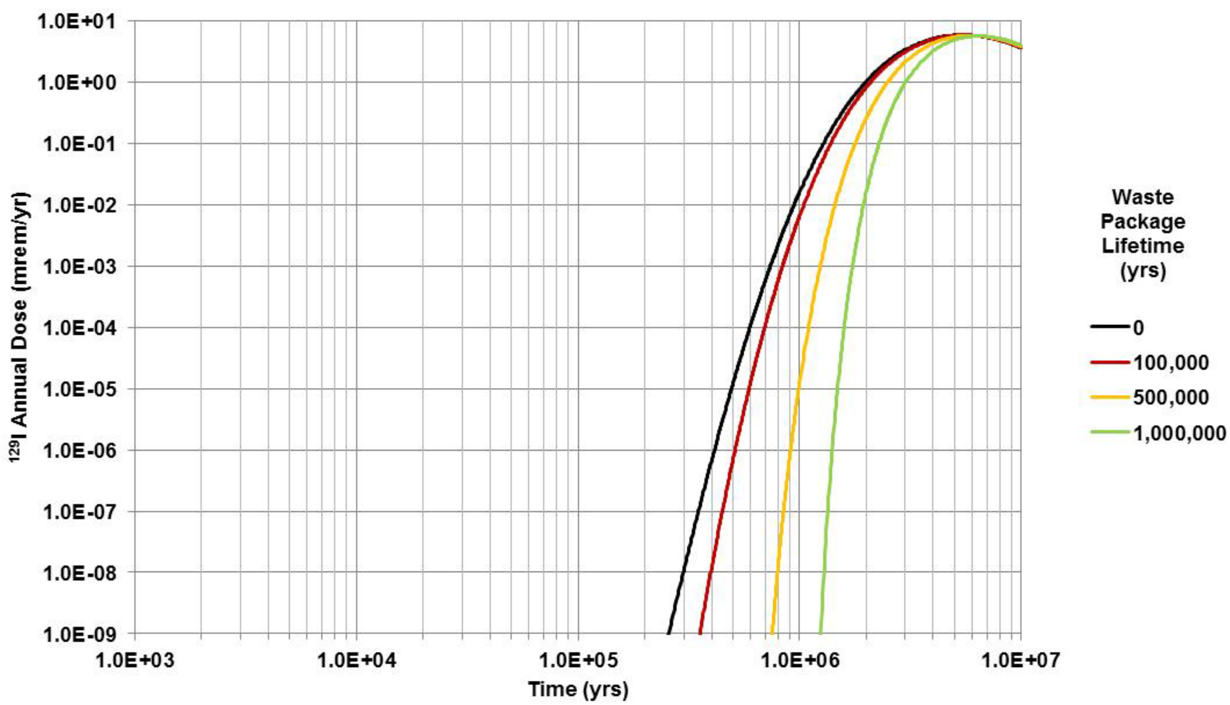


Figure 4-17. Effect of Waste Package Lifetime on Annual Dose from ^{129}I in the Clay GDS Model `

Bentonite Buffer Integrity—The effect of the integrity of the bentonite buffer and DRZ clay on ^{129}I annual dose is shown in Figure 4-18. The sensitivity analysis includes two cases:

- Baseline Intact Bentonite and DRZ—The bentonite buffer is 1.025-m thick and the fissured clay DRZ is 1.15-m thick for a total thickness of 2.175 m. There are no significant fractures through the buffer or DRZ. A constant darcy velocity through the engineered barrier system and far field of 6.3×10^{-7} m/yr is assumed. This flow velocity is low enough that transport through the engineered barrier system and far field is diffusion dominated.
- Damaged Bentonite and DRZ—The thickness of the bentonite buffer and the fissured clay DRZ are both reduced by a factor is 5, for a total effective thickness of 0.435 m. A multiplier of 1,000 is applied to the flow velocity in the engineered barrier system, resulting in advection-dominated transport in the engineered barrier system. These enhanced transport properties are considered to represent the effects of non-healing fractures connecting the repository and the far-field clay.

The damaged buffer has little effect on annual dose because the combined buffer and DRZ thickness of 2.175 m is much less than the overlying clay thickness of 150 m. Enhanced transport through the engineered barrier system is attenuated by slow diffusive transport in the far-field clay.

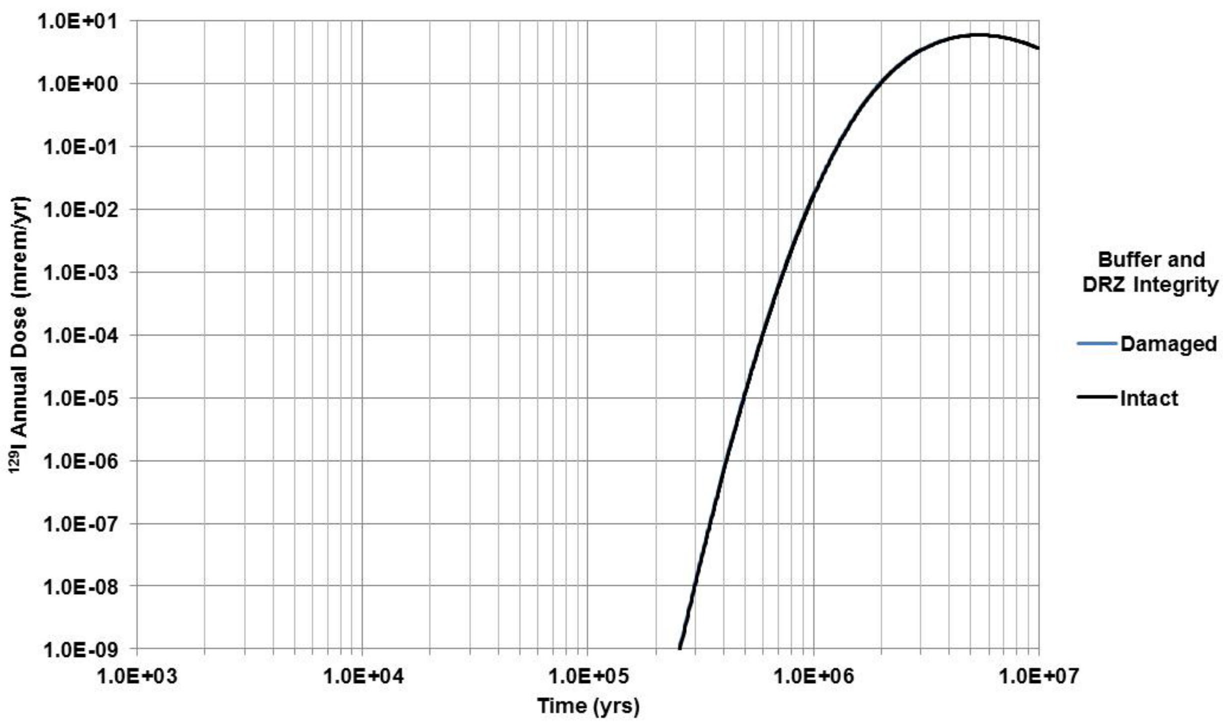


Figure 4-18. Effect of Buffer and DRZ Integrity on Annual Dose from ^{129}I in the Clay GDS Model

Flow Rate—The effect of the flow rate on ^{129}I annual dose is shown in Figure 4-19. The sensitivity analysis includes three cases:

- Increased Flow Rate (6.3×10^{-6} m/yr)—The baseline darcy velocity through the buffer, DRZ, and far-field clay is multiplied by a factor of 10. This flow velocity results in advection-dominated transport at certain times and locations within the disposal system.
- Baseline Flow Rate (6.3×10^{-7} m/yr)—The darcy velocity through the buffer, DRZ and far-field clay is 6.3×10^{-7} m/yr. This flow velocity results in diffusion-dominated transport throughout the disposal system.
- Decreased Flow Rate (6.3×10^{-8} m/yr)—The baseline darcy velocity through the buffer, DRZ, and far-field clay is reduced by a factor of 10. This flow velocity results in diffusion-dominated transport throughout the disposal system.

The case where the brine flow rates are increased has a much more significant effect on dose than when the flow rates are decreased. This is because the factor-of-10 increase results in advection-dominated transport at certain times and locations within the disposal system. The effect of the advective transport is to increase the magnitude of the peak dose by about a factor of 10 and accelerate the time of the peak dose by about a factor of 3.

The effects of the factor-of-10 brine flow decrease are much smaller (only about a factor of 2 decrease in peak dose) because the decrease in flow is only decreasing the contribution from advective transport, which is already smaller than the contribution from diffusive transport.

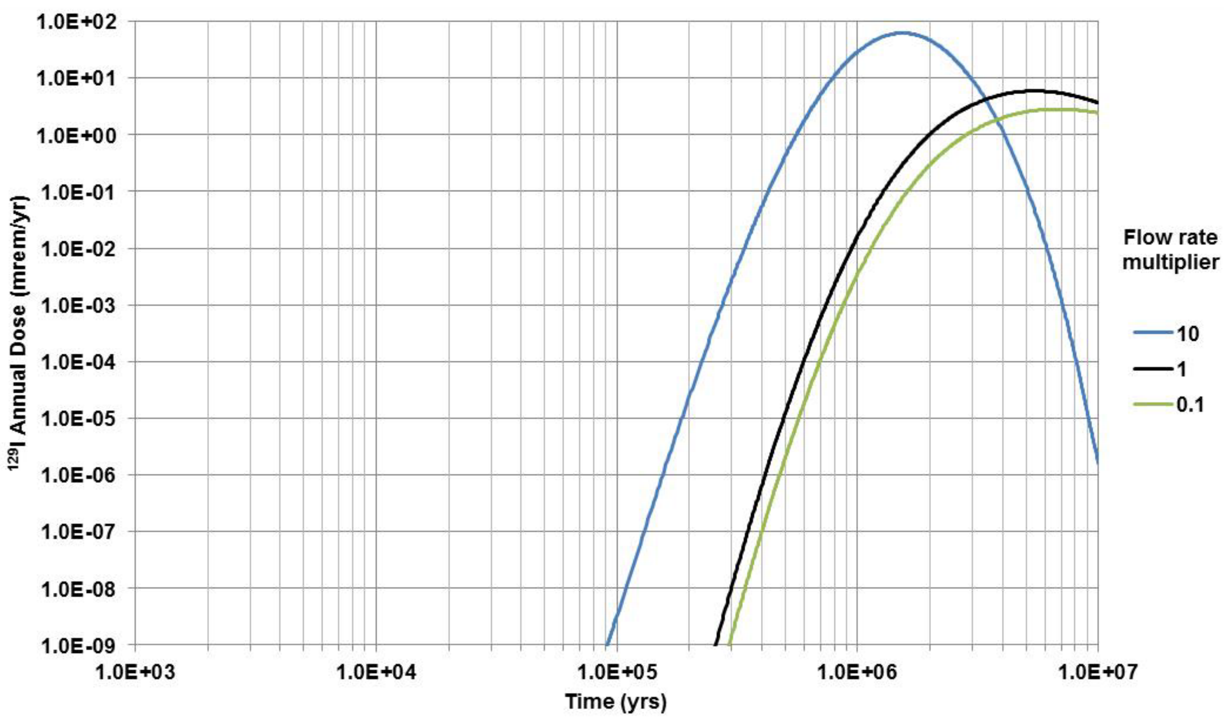


Figure 4-19. Effect of Flow Rate on Annual Dose from ^{129}I in the Clay GDS Model

Diffusion—The effect of the diffusion coefficient on ^{129}I annual dose is shown in Figure 4-20. The sensitivity analysis includes two cases:

- Baseline Molecular (Free Water) Diffusion Coefficient (2.30×10^{-9} m^2/s)—The corresponding effective diffusion coefficient for ^{129}I in each region is based on the local porosity and tortuosity.
- Enhanced Molecular (Free Water) Diffusion Coefficient (1.15×10^{-8} m^2/s)—This results in a corresponding effective diffusion coefficient for ^{129}I in each region that is a factor of 5 larger than the baseline value. The increased molecular diffusion coefficient reproduces potential changes in the effective diffusion coefficient for ^{129}I that might result from changes in available porosity (e.g., due to anion exclusion) or tortuosity.

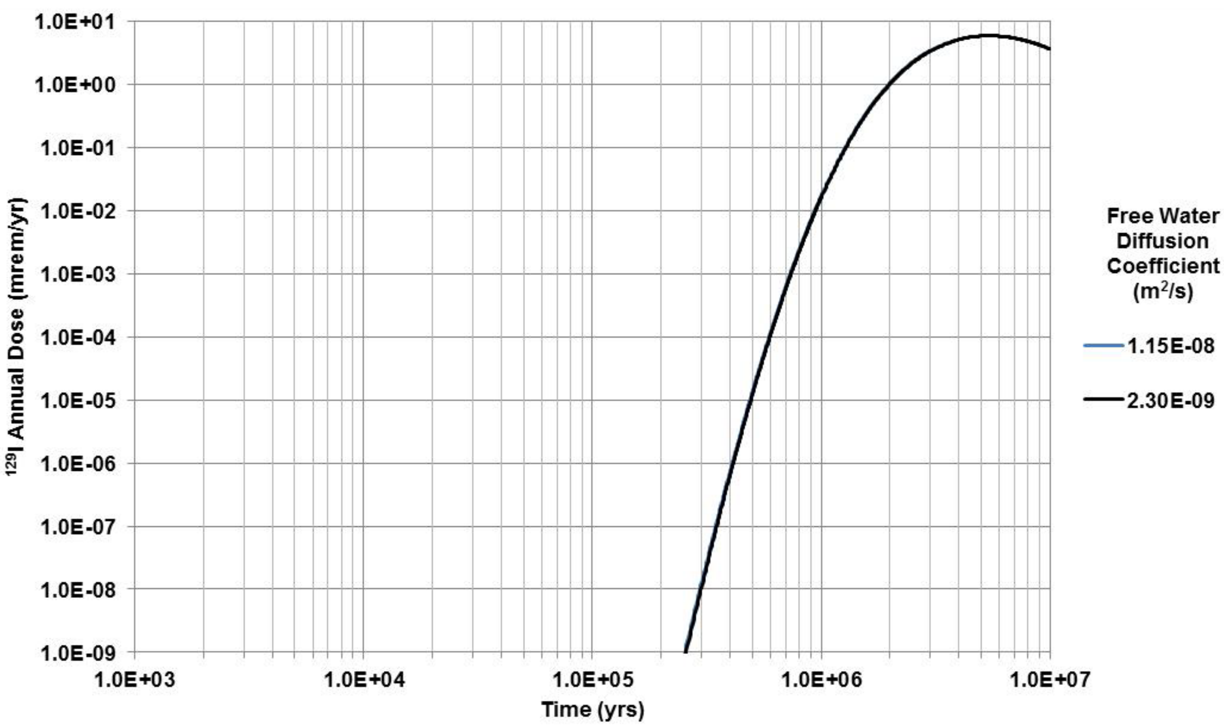


Figure 4-20. Effect of Diffusion Coefficient on Annual Dose from ^{129}I in the Clay GDS Model

In the high diffusion coefficient case, diffusive fluxes would be expected to be a factor of five higher than in the baseline case. However, the effect of the high diffusion coefficient on annual dose is not significant. This is because, in the diffusive-transport-dominated clay disposal system, the effects of a higher diffusive flux on transport through the buffer and DRZ are attenuated in the 150-m, two-dimensional, far-field clay by (1) the conceptual model assumptions about the system geometry, and (2) lateral diffusion in the far-field clay.

At the scale of a single waste package, diffusion into the far-field clay occurs from a single DRZ cell (grid block) into a single clay inlet cell (grid block) across a 79-m^2 diffusion area, corresponding to the surface area of a cylindrical engineered barrier system (bentonite buffer and clay DRZ) around a waste package. Diffusion within the far-field clay then occurs from the single inlet cell both vertically and horizontally through a 20 × 20 two-dimensional network of cells, where the vertical diffusion area in a single cell is 1.2375 m^2 and the horizontal diffusion area is 33.75 m^2. Diffusive transport to the receptor location is in the vertical direction, through 150 m of clay. Since diffusion into the far-field clay all enters a single cell and the diffusion area in is greater than the diffusion area out, the single clay inlet cell tends to attenuate diffusive transport. Furthermore, for the mass that does diffuse out of the clay inlet cell, horizontal (lateral) diffusion into the rest of the far-field clay tends to be much greater than vertical (longitudinal) diffusion, due to the larger diffusive area in the horizontal direction. Therefore, the higher diffusive flux associated with the high diffusion coefficient is offset by attenuation in the far-field clay inlet cell and by lateral diffusion in the far-field clay.

Far-Field Sorption—The effect of sorption in the far-field clay on ^{129}I annual dose is shown in Figure 4-21. The sensitivity analysis includes four cases:

- Baseline ^{129}I Sorption (k_d = 0.00 mL/g)—The corresponding retardation factor is 1.0.
- Increased ^{129}I Sorption (k_d = 0.01 mL/g)—The corresponding retardation factor is 1.1.
- Increased ^{129}I Sorption (k_d = 0.10 mL/g)—The corresponding retardation factor is 2.1.
- Increased ^{129}I Sorption (k_d = 1.0 mL/g)—The corresponding retardation factor is 12.1.

Changes in ^{129}I k_d have a significant effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. For the case with k_d = 0.10 mL/g and R_f = 2.1, the dose curve shifts to the right by a factor of 2.1 on the time axis. For the case with k_d = 1.0 mL/g and R_f = 12.1, the dose curve shifts to the right by a factor of 12.1 on the time axis.

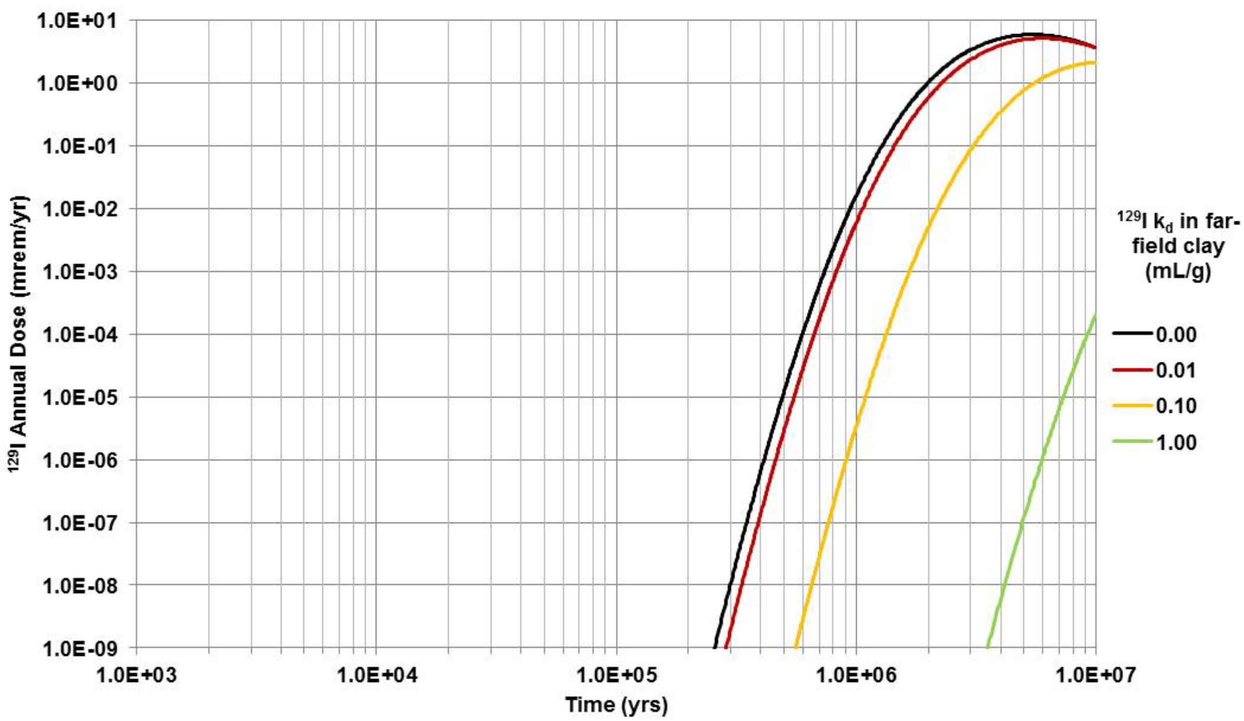


Figure 4-21. Effect of Clay Sorption on Annual Dose from ^{129}I in the Clay GDS Model

Clay Thickness (Receptor Distance)—The effect of far-field clay thickness overlying the emplaced waste on ^{129}I annual dose is shown in Figure 4-22. The sensitivity analysis includes three cases:

- Reduced overlying clay thickness (75 m)
- Baseline overlying clay thickness (150 m)
- Increased overlying clay thickness (200 m)

The annual dose is quite sensitive to the thickness of the overlying far-field clay, which represents the effective distance to the receptor location. In these clay disposal system simulations, the effects of reducing the overlying clay thickness on dose are approximately linear.

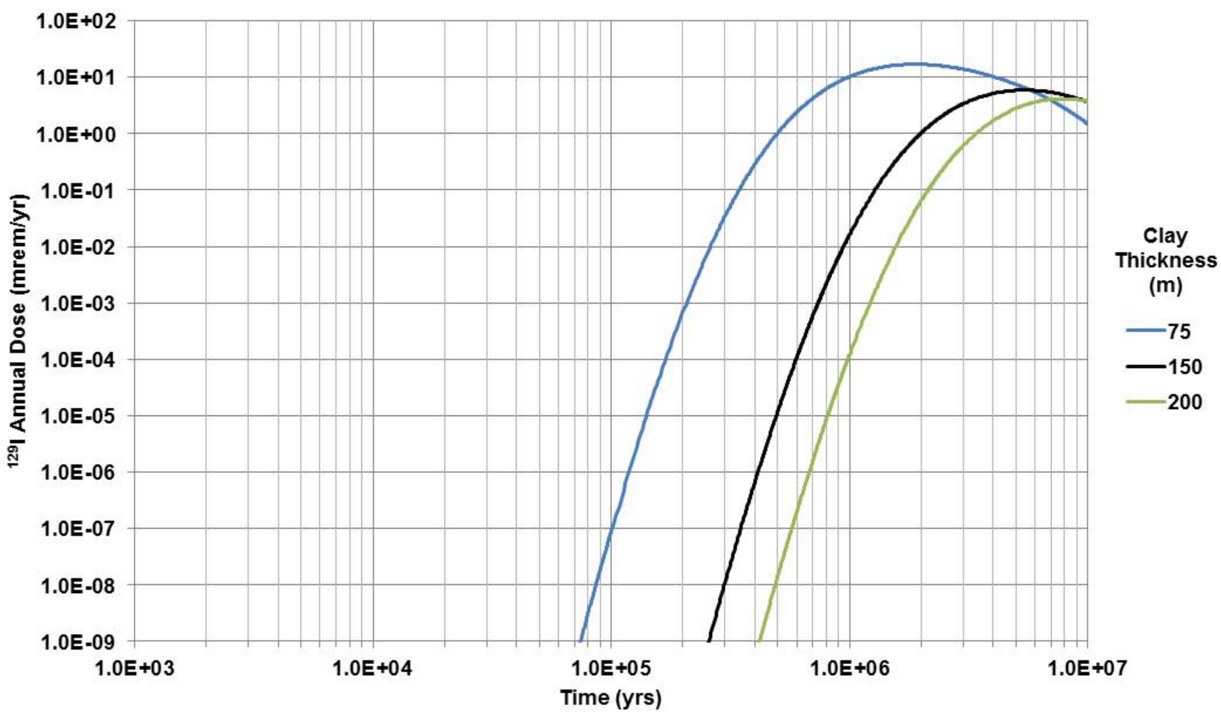


Figure 4-22. Effect of Overlying Clay Thickness on Annual Dose from ^{129}I in the Clay GDS Model

Summary—Based on these seven sensitivity analyses examining "one-off" conditions from the baseline scenario, the following observations can be made regarding the performance of a generic clay disposal system under baseline scenario conditions:

- Processes and parameters affecting radionuclide transport through the overlying 150-m far-field clay can have a significant effect on annual dose. These include sorption, k_d, and clay thickness (distance to receptor).
- Processes and parameters affecting radionuclide transport through the entire clay disposal system can have a significant effect on annual dose. Increasing the flow rate to produce advection-dominated transport increases the dose. This system-wide effect is most important in the far-field clay.

- Processes and parameters affecting waste form degradation can have a moderate effect on annual dose. Increasing the degradation rate does not significantly increase the dose because the effects are mitigated by slow diffusion through the far-field clay. Decreasing the degradation rate decreases the annual dose.
- Processes and parameters affecting waste package lifetime can have a moderate effect on annual dose. Increasing the waste package lifetime delays the onset of waste form degradation and radionuclide release from the waste form.
- Processes and parameters affecting radionuclide transport through the 2.175-m engineered barrier system (bentonite buffer and fissured clay DRZ) have a minimal effect on dose.

4.4.2.3 Granite GDS Model Sensitivity Analyses

Annual dose results from the deterministic granite GDS baseline scenario are shown in Figure 4-6. The following one-off sensitivity simulations were performed to investigate the effects on ^{129}I movement through the disposal system:

- Waste form fractional degradation rate and gap fraction (Figure 4-23)
- Waste package lifetime (Figure 4-24)
- Sorption (^{129}I distribution coefficient) in the bentonite buffer (Figure 4-25)
- Flow rate in the near-field and far-field granite (Figure 4-26)
- Molecular diffusion coefficient (Figure 4-27)
- Sorption (^{129}I distribution coefficient) in the far-field granite (Figure 4-28)
- Fracture spacing in the far-field granite (Figure 4-29)
- Distance to receptor (Figure 4-30)

Waste Form Degradation—The effect of waste form degradation rate on ^{129}I annual dose is shown in Figure 4-23. The sensitivity analysis includes four fractional degradation rate cases:

- <u>Fast Waste Form Degradation (0.1 yr^{-1})</u>—100% of the radionuclide mass is released from the waste form in the first 250 years. This provides a bounding case for instantaneous release of gap and grain boundary inventory from the waste form equivalent to a gap fraction of 1.0.
- <u>Baseline Waste Form Degradation (2×10^{-5} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form in the first 35,000 years, 95% of the mass is released by 150,000 years, and 99.9% of the mass is released by about 350,000 years.
- <u>Baseline Waste Form Degradation (2×10^{-5} yr^{-1}) with 0.2675 gap fraction</u>—The same fractional degradation rate as the baseline case, but 26.75% of the initial ^{129}I mass is released instantaneously. This corresponds to the maximum ^{129}I gap fraction in Sandia National Laboratories (2008c, Table 6.3.7-29).
- <u>Slow Waste Form Degradation (1×10^{-7} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form after 4,800,000 years, and 76% of the mass is released by 10,000,000 years.

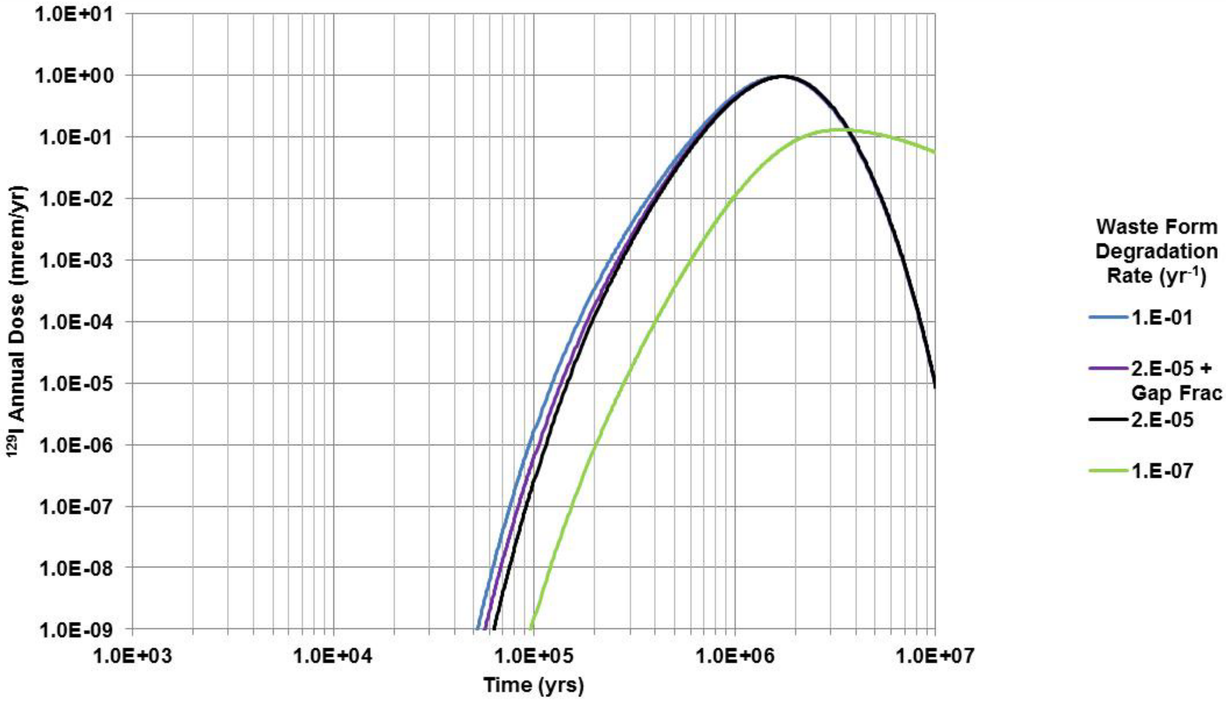


Figure 4-23. Effect of Waste Form Degradation Rate and Gap Fraction on Annual Dose from ^{129}I in the Granite GDS Model

In the fast degradation rate (0.1 yr^{-1}) case, all mass is released (degraded) from the waste form in the first 250 years. This is roughly equivalent to assuming that all ^{129}I mass is instantaneously released as gap and grain boundary inventory (i.e., a ^{129}I gap fraction of 1.0 as compared to the estimated range of 0.0204–0.2675). The mass released from the waste form diffuses vertically through the 0.36-m thick bentonite buffer, and then advects through the 0.78-m thick granite DRZ and 5,000 m of far-field fractured granite. In the fast degradation rate case, 44% of the initial ^{129}I mass reaches the granite by 100,000 years, whereas in the baseline case, only 37% of the initial mass reaches the granite by 100,000 years, and 14% of the mass is still undegraded. Despite the greater early transport of ^{129}I mass away from the repository in the fast degradation rate case, the effect on annual dose is not significant. This is because the effect of the early mass on annual dose is offset by (1) diffusion-dominated transport in the bentonite buffer which tends to attenuate the releases, and (2) long travel times through the far-field granite.

The result from the case with the baseline degradation rate and a 0.2675 gap fraction falls between the baseline result (gap fraction of 0.0) and the fast degradation rate case result (gap fraction of 1.0). This further emphasizes that the gap fraction does not have a significant effect under the granite baseline scenario assumptions.

For the slow fractional degradation rate (1×10^{-7} yr^{-1}), 50% of the radionuclide mass is not released from the waste form until 4,800,000 years, and only 76% of the mass is released by 10,000,000 years. In this case, the degradation time is slower than the travel time through the far-field granite and the effect of the slower degradation rate is to reduce the magnitude of the peak dose by about a factor of 7 and delay the time of the peak dose by about a factor of 2.

Waste Package Degradation—The effect of waste package lifetime on ^{129}I annual dose is shown in Figure 4-24. The sensitivity analysis includes four cases:

- Baseline Waste Package Lifetime (0 years)—1% (160) of the waste packages fail instantaneously.
- Moderate Waste Package Lifetime (100,000 years)—160 waste packages fail at 100,000 years.
- Long Waste Package Lifetime (500,000 years)—160 waste packages fail at 500,000 years.
- Very Long Waste Package Lifetime (1,000,000 years)—160 waste packages fail at 1,000,000 years.

The effect of waste package lifetime on system performance is to delay the onset of waste form degradation and radionuclide release from the waste form. The delay is evident in the annual dose curves; they are all shifted to the right on the time axis (100,000, 500,000, and 1,000,000 years) relative to the baseline case.

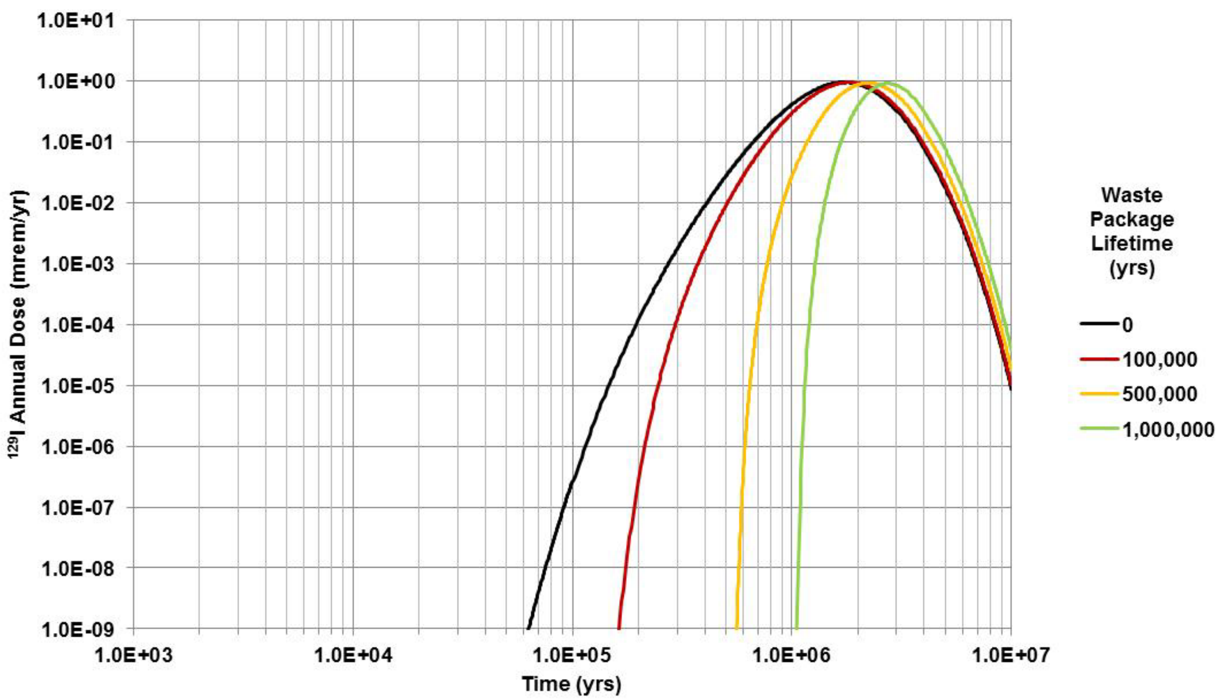


Figure 4-24. Effect of Waste Package Lifetime on Annual Dose from ^{129}I in the Granite GDS Model

Bentonite Sorption—The effect of sorption in the bentonite buffer on ^{129}I annual dose is shown in Figure 4-25. The sensitivity analysis includes three cases:

- Baseline ^{129}I Sorption (k_d = 0.0 mL/g)—The corresponding retardation factor is 1.0.
- Increased ^{129}I Sorption (k_d = 1.0 mL/g)—The corresponding retardation factor is 4.6.
- Increased ^{129}I Sorption (k_d = 5.0 mL/g)**—**The corresponding retardation factor is 19.0.

Changes in ^{129}I k_d in the bentonite buffer have a moderate effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. However, due to the small (0.36 m) transport length of the buffer relative to the 5,000-m transport length of the far-field granite, sorption in the bentonite buffer is not as important to overall system performance as measured by annual dose. For the case with k_d = 5.0 mL/g and R_f = 19.0, the dose curve only shifts to the right by a factor of about 1.2 on the time axis relative to the baseline case.

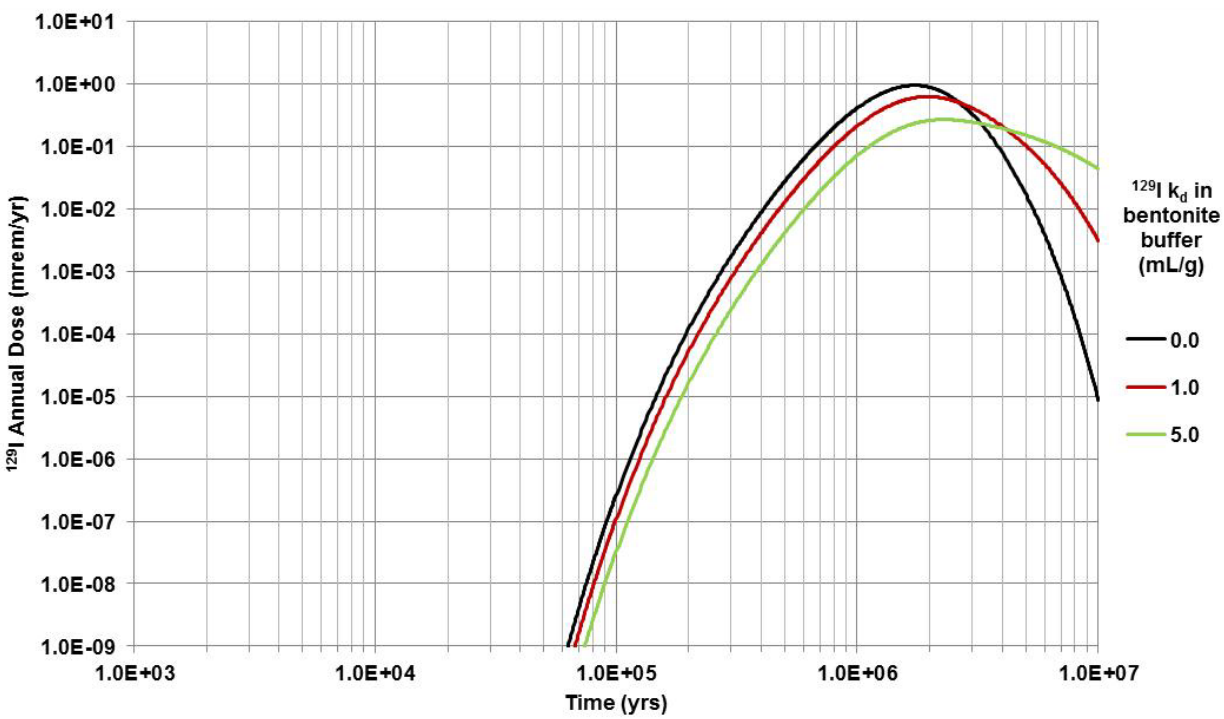


Figure 4-25. Effect of Sorption in Bentonite Buffer on Annual Dose from ^{129}I in the Granite GDS Model

Flow Rate—The effect of the flow rate on ^{129}I annual dose is shown in Figure 4-26. The sensitivity analysis includes three cases:

- Increased Volumetric Flow Rate (5.1×10^{-3} m^3/yr)—The baseline volumetric flow rate and darcy velocity through the granite is increased by a factor of 10. This increased flow velocity results in advection-dominated transport through the granite.
- Baseline Volumetric Flow Rate (5.1×10^{-4} m^3/yr)—The corresponding darcy velocity through the granite is 9.6×10^{-6} m/yr. This flow velocity results in advection-dominated transport through the granite.
- Decreased Volumetric Flow Rate (5.1 x 10^{-5} m^3/yr)—The baseline volumetric flow rate and darcy velocity through the granite is reduced by a factor of 10. This reduced flow velocity still results in advection-dominated transport through the granite.

In the advection-dominated granite disposal system, the effect of changing the flow rate is to shift the time of peak dose by a corresponding factor along the time axis. The magnitude of peak is lower with lower flow rates due to greater radioactive decay and greater dispersion as the peak moves further out in time.

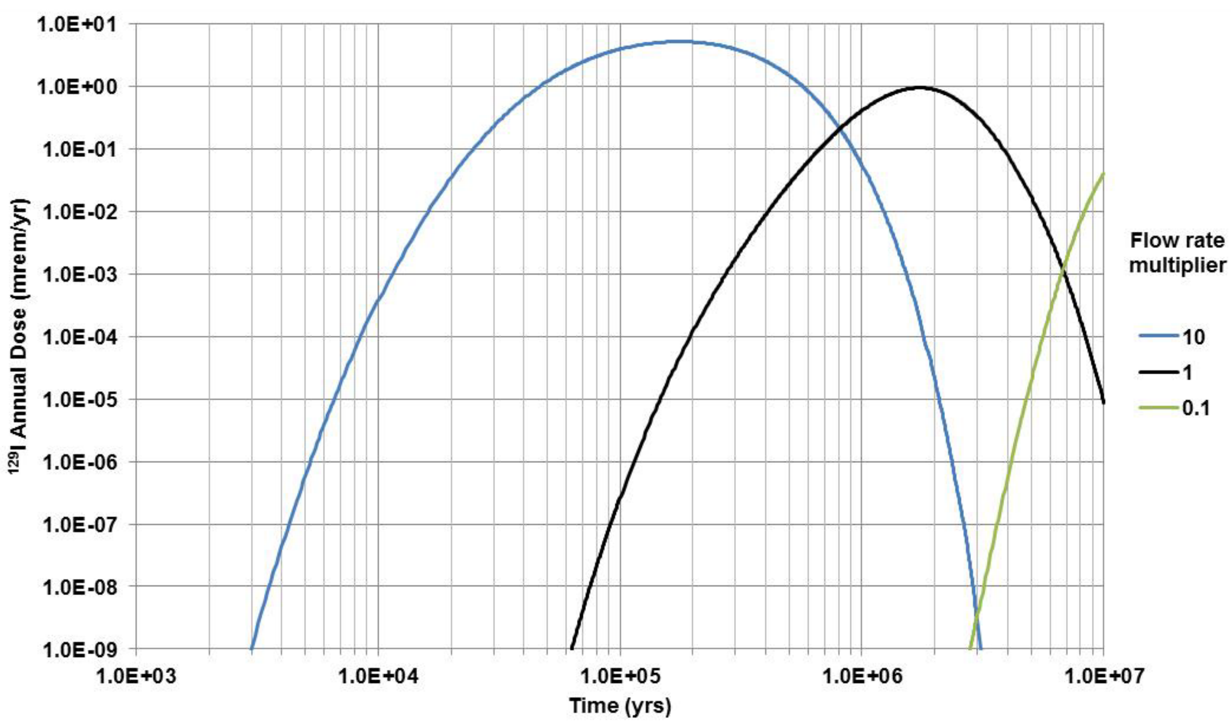


Figure 4-26. Effect of Flow Rate on Annual Dose from ^{129}I in the Granite GDS Model

Diffusion—The effect of the diffusion coefficient on ^{129}I annual dose is shown in Figure 4-27. The sensitivity analysis includes two cases:

- Baseline Molecular (Free Water) Diffusion Coefficient (2.30×10^{-9} m^2/s)—The corresponding effective diffusion coefficient for ^{129}I in each region is based on the local porosity and tortuosity.
- Enhanced Molecular (Free Water) Diffusion Coefficient (1.15×10^{-8} m^2/s)—This results in a corresponding effective diffusion coefficient for ^{129}I in each region that is a factor of 5 larger than the baseline value. The increased molecular diffusion coefficient reproduces potential changes in the effective diffusion coefficient for ^{129}I that might result from changes in available porosity (e.g., due to anion exclusion) or tortuosity.

Increasing the diffusion coefficient has no effect on the annual dose in an advection-dominated system.

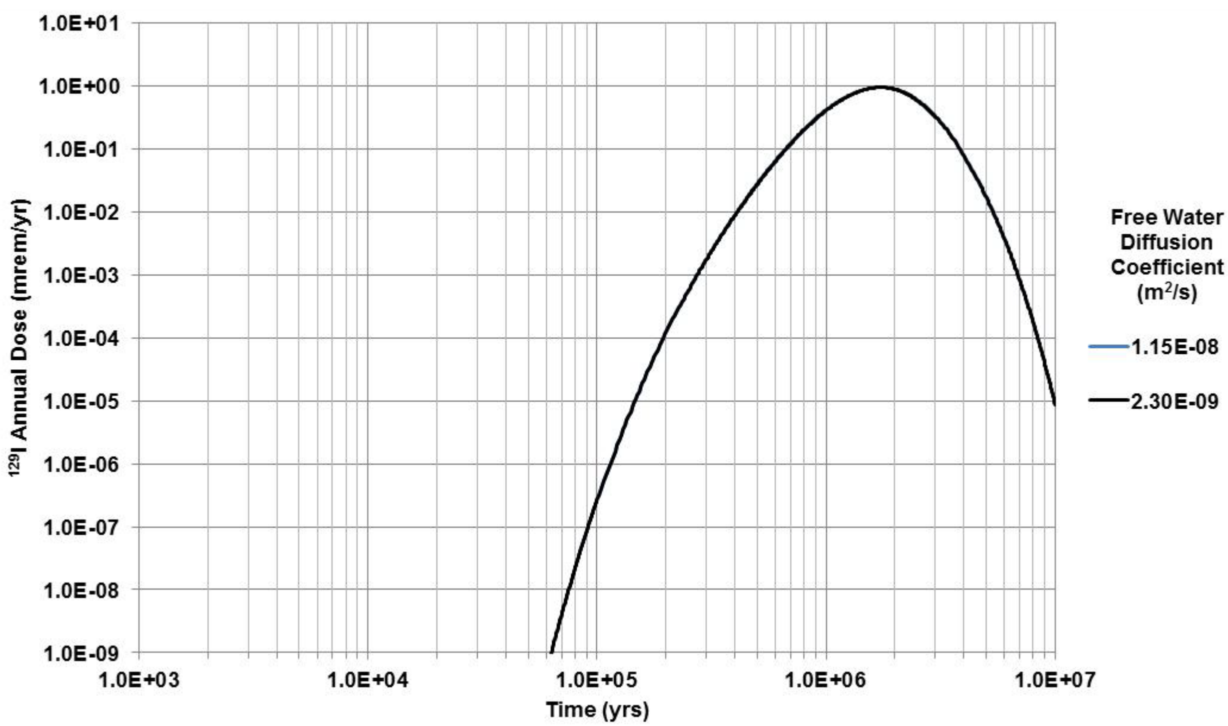


Figure 4-27. Effect of Diffusion Coefficient on Annual Dose from ^{129}I in the Granite GDS Model

Far-Field Sorption—The effect of sorption in the far-field granite on ^{129}I annual dose is shown in Figure 4-28. The sensitivity analysis includes three cases:

- Baseline ^{129}I Sorption (k_d = 0.00 mL/g)—The corresponding retardation factor is 1.0.
- Increased ^{129}I Sorption (k_d = 0.01 mL/g)—The corresponding retardation factor is 16.0.
- Increased ^{129}I Sorption (k_d = 0.10 mL/g)—The corresponding retardation factor is 151.

Changes in ^{129}I k_d in the far field have a significant effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. For the case with k_d = 0.01 mL/g and R_f = 16.0, the dose curve shifts to the right by a factor of 16 on the time axis. For the case with k_d = 0.10 mL/g and R_f = 151, the dose curve shifts so far to the right on the time axis that it does not show on the plot.

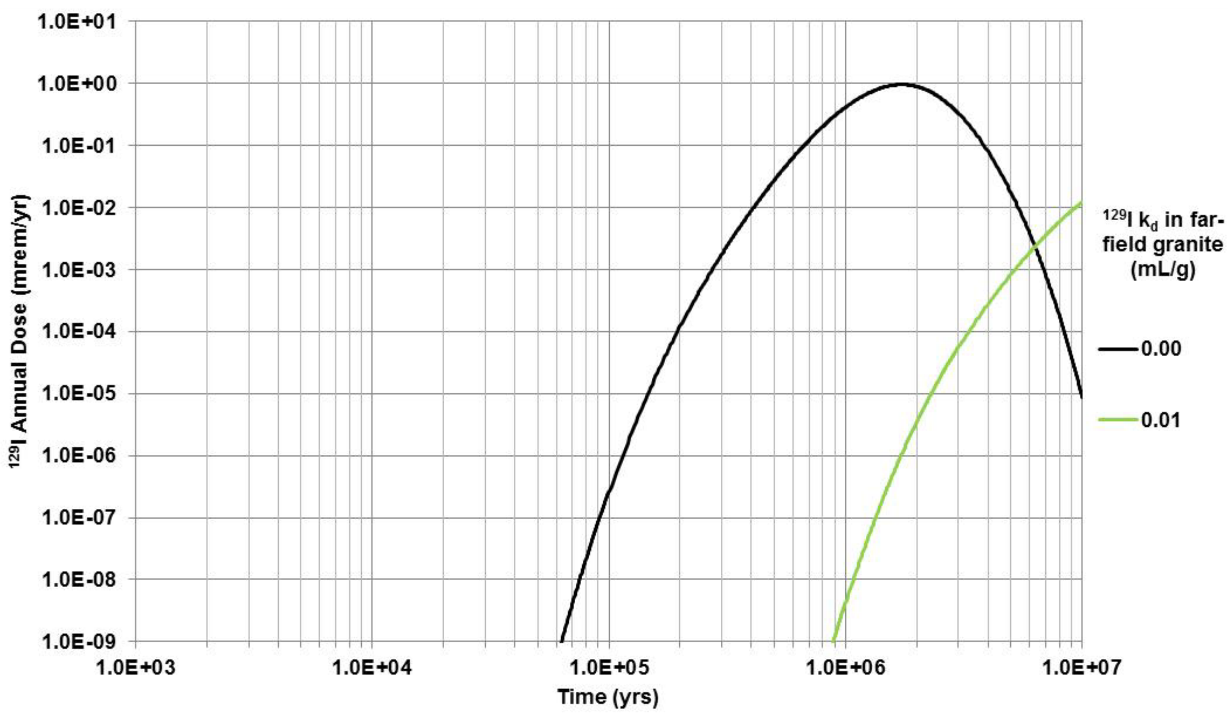


Figure 4-28. Effect of Sorption in Granite on Annual Dose from ^{129}I in the Granite GDS Model

Fracture Spacing—The effect of fracture spacing in the far-field granite on ^{129}I annual dose is shown in Figure 4-29. The sensitivity analysis includes two cases:

- Baseline fracture spacing (25 m)
- Reduced fracture spacing (10 m)

Fracture spacing in the granite affects the matrix diffusion length. The baseline case with a larger matrix diffusion length produces more matrix diffusion and a corresponding greater delay in advective transport through the fracture.

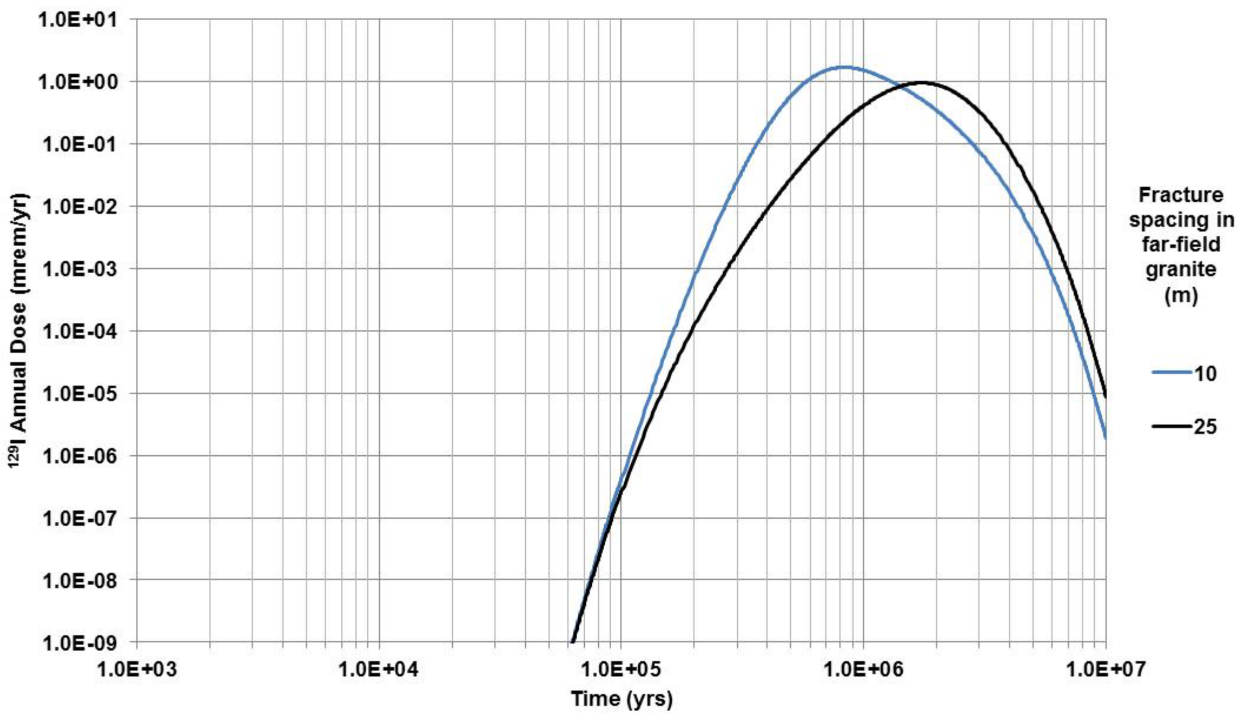


Figure 4-29. Effect of Fracture Spacing in Granite on Annual Dose from ^{129}I in the Granite GDS Model

Receptor Distance—The effect of distance to the receptor on ^{129}I annual dose is shown in Figure 4-30. The sensitivity analysis includes three cases:

- Baseline granite length to receptor (5,000 m)
- Reduced granite length to receptor (3,000 m)
- Reduced granite length to receptor (1,000 m)

The annual dose is quite sensitive to the granite length to the receptor location. In this advection dominated granite disposal system simulations, the effects of reducing the granite length are approximately linear.

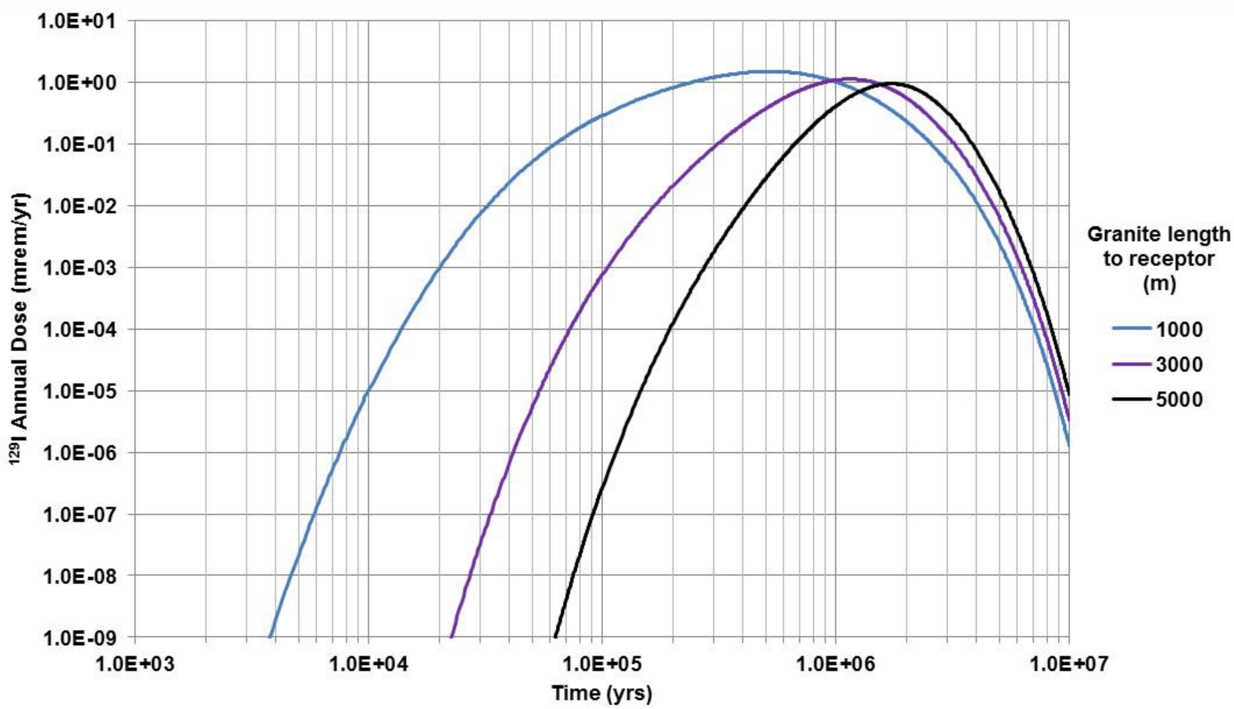


Figure 4-30. Effect of Distance to Receptor on Annual Dose from ^{129}I in the Granite GDS Model

Summary—Based on these eight sensitivity analyses examining "one-off" conditions from the baseline scenario, the following observations can be made regarding the performance of a generic granite disposal system under baseline scenario conditions:

- Processes and parameters affecting radionuclide transport through the 5,000-m far-field granite can have a significant effect on annual dose. These include sorption, k_d, distance to receptor, and fracture spacing.
- Processes and parameters affecting radionuclide transport through the entire granite disposal system can have a significant effect on annual dose. Increasing or decreasing the flow rate correspondingly affects the dose. This system-wide effect is most important in the advection-dominated far-field granite.
- Processes and parameters affecting waste form degradation can have a moderate effect on annual dose. Increasing the degradation rate does not significantly increase the dose because the effects are mitigated by slow diffusion through the bentonite buffer and a long travel time through the far-field granite. Decreasing the degradation rate decreases the annual dose.
- Processes and parameters affecting waste package lifetime can have a significant effect on annual dose. Increasing the waste package lifetime delays the onset of waste form degradation and radionuclide release from the waste form.
- Processes and parameters affecting radionuclide transport through the 0.36-m bentonite buffer can have a moderate effect on dose. This includes sorption, k_d.

4.4.2.4 Deep Borehole GDS Model Sensitivity Analyses

Annual dose results from the deterministic deep borehole GDS baseline scenario are shown in Figure 4-8. The following one-off sensitivity simulations were performed to investigate the effects on ^{129}I movement through the disposal system:

- Waste form fractional degradation rate (Figure 4-31)
- Sorption (^{129}I distribution coefficient) in the disposal zone (Figure 4-32)
- Sorption (^{129}I distribution coefficient) in the seal zone (Figure 4-33)
- Molecular diffusion coefficient (Figure 4-34)

In addition, the effects of seal zone integrity and flow rate up the borehole are discussed.

Waste Form Degradation—The effect of waste form degradation rate on ^{129}I annual dose is shown in Figure 4-31. The sensitivity analysis includes the same three fractional degradation rate cases described in Sections 4.4.2.1 and 4.4.2.2:

- <u>Fast Waste Form Degradation (0.1 yr^{-1})</u>—100% of the radionuclide mass is released from the waste form in the first 250 years. This provides a bounding case for instantaneous release of gap and grain boundary inventory from the waste form.
- <u>Baseline Waste Form Degradation (2×10^{-5} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form in the first 35,000 years, 95% of the mass is released by 150,000 years, and 99.9% of the mass is released by about 350,000 years.
- <u>Slow Waste Form Degradation (1×10^{-7} yr^{-1})</u>—50% of the radionuclide mass is released from the waste form after 4,800,000 years, and 76% of the mass is released by 10,000,000 years.

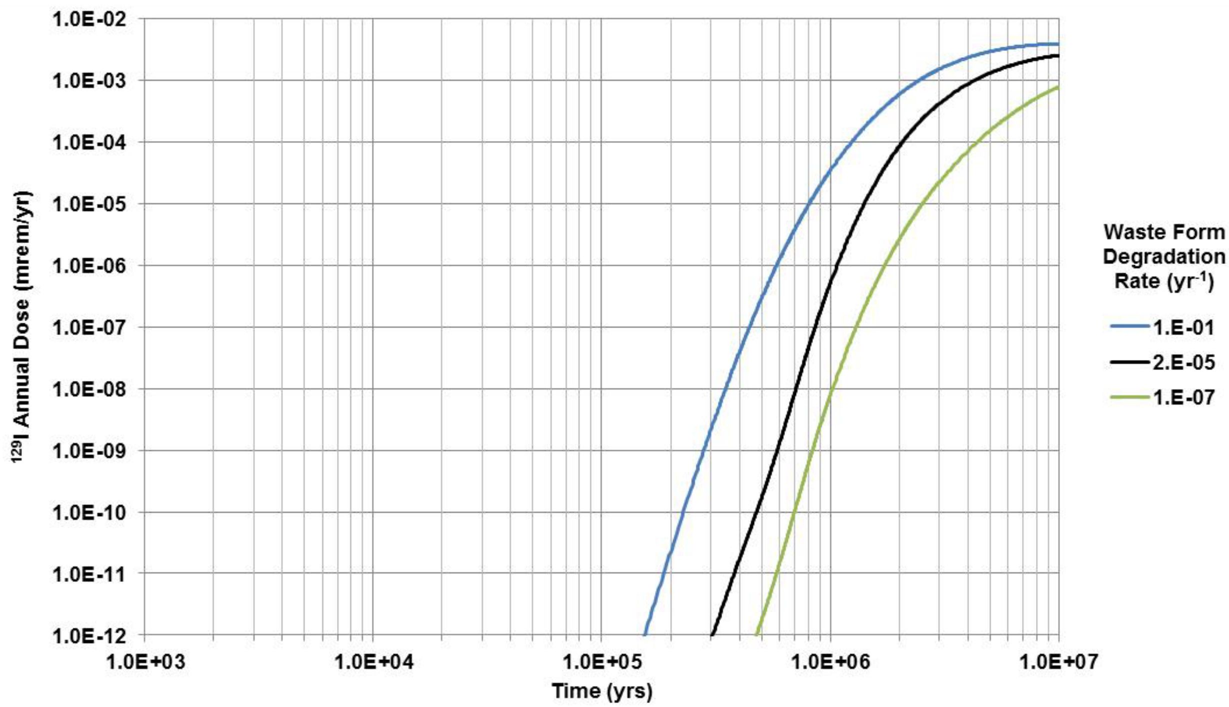


Figure 4-31. Effect of Waste Form Degradation Rate on Annual Dose from ^{129}I in the Deep Borehole GDS Model

In the fast degradation rate (0.1 yr^{-1}) case, all mass is released (degraded) from the waste form in the first 250 years. This is roughly equivalent to assuming that all ^{129}I mass is instantaneously released as gap and grain boundary inventory (i.e., a ^{129}I gap fraction of 1.0 as compared to the estimated range of 0.0204–0.2675). The mass released from the waste forms advects upward through the 2,000-m disposal zone and then diffuses upward through the 1,000-m seal zone. However, the relative contributions of advective and diffusive transport vary with time and distance up the borehole (flow rates decrease with time and with distance up the borehole). In the fast degradation rate case, 23% of the initial ^{129}I mass reaches the seal zone by 100,000 years, whereas in the baseline case, only 11% of the initial mass reaches the seal zone by 100,000 years, and 22% of the mass is still undegraded. Despite the greater early transport of ^{129}I mass away from the repository in the fast degradation rate case, the effect on annual dose is only moderate. This is because some of the effect of the early mass on annual dose is offset by diffusion-dominated transport in the upper part of the seal zone which tends to attenuate the releases.

For the slow fractional degradation rate (1×10^{-7} yr^{-1}), 50% of the radionuclide mass is not released from the waste form until 4,800,000 years, and only 76% of the mass is released by 10,000,000 years. In this case, the slow degradation time means that a smaller fraction of the released mass is available for transport during early time when advective transport is more predominant. As a result, the annual dose is lower than for the baseline case.

Disposal Zone Sorption—The effect of sorption in the disposal zone on ^{129}I annual dose is shown in Figure 4-32. The sensitivity analysis includes four cases:

- Baseline ^{129}I Sorption (k_d = 0.00 mL/g)—The corresponding retardation factor is 1.0.
- Increased ^{129}I Sorption (k_d = 0.01 mL/g)—The corresponding retardation factor is 1.7.
- Increased ^{129}I Sorption (k_d = 0.10 mL/g)—The corresponding retardation factor is 8.2.
- Increased ^{129}I Sorption (k_d = 1.0 mL/g)—The corresponding retardation factor is 73.2.

Changes in ^{129}I k_d in the disposal zone have a moderate effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. However, sorption in the disposal zone is not as important to overall system performance as sorption in the seal zone, because transport in the disposal zone is advection-dominated over a greater length for a longer period of time. For the case with disposal zone k_d = 0.1 mL/g and R_f = 8.2, the dose curve only shifts to the right by a factor of about 1.2 on the time axis relative to the baseline case.

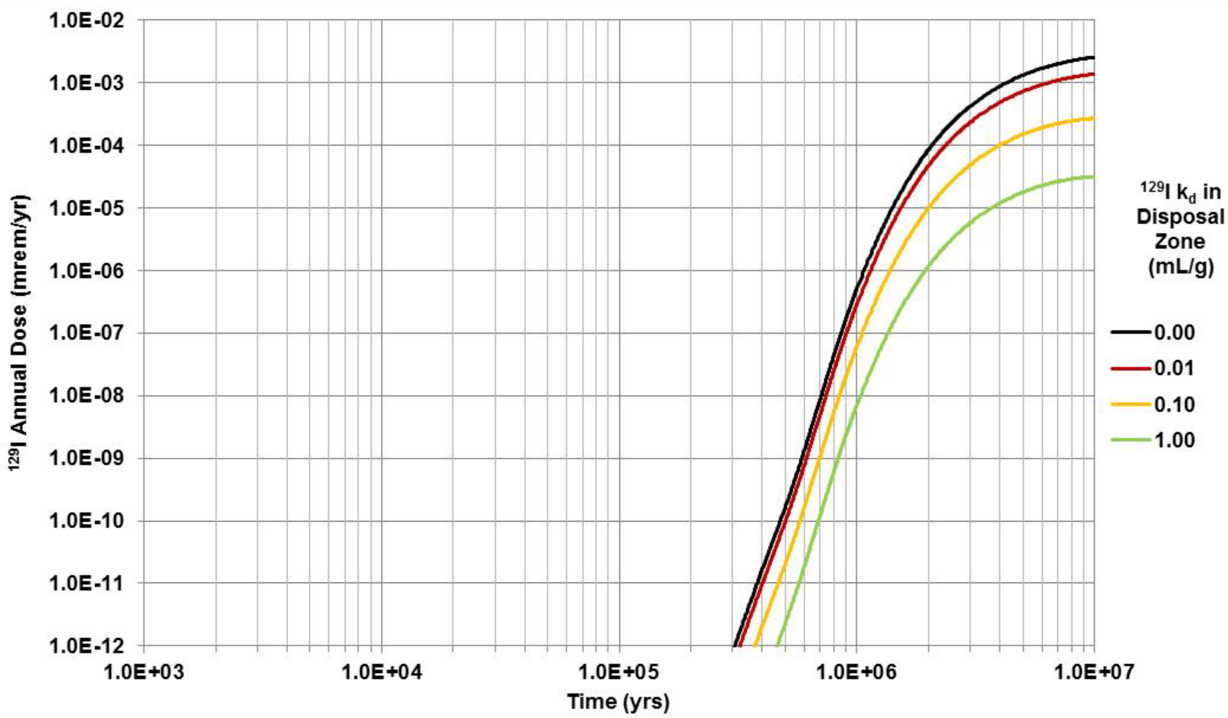


Figure 4-32. Effect of Sorption in the Disposal Zone on Annual Dose from ^{129}I in the Deep Borehole GDS Model

Seal Zone Sorption—The effect of sorption in the seal zone on ^{129}I annual dose is shown in Figure 4-33. The sensitivity analysis includes four cases:

- Baseline ^{129}I Sorption (k_d = 0.00 mL/g)—The corresponding retardation factor is 1.0.
- Increased ^{129}I Sorption (k_d = 0.01 mL/g)—The corresponding retardation factor is 1.7.
- Increased ^{129}I Sorption (k_d = 0.10 mL/g)—The corresponding retardation factor is 8.2.
- Increased ^{129}I Sorption (k_d = 1.0 mL/g)—The corresponding retardation factor is 73.2.

Changes in ^{129}I k_d have a significant effect on annual dose. This is because of the delay in transport that is represented by the associated retardation factor. Because transport through the seal zone is the slowest component in the deep borehole disposal system, the effect of increased seal zone k_d is to shift the dose curves to the right on the time axis by a factor that corresponds to the retardation factor.

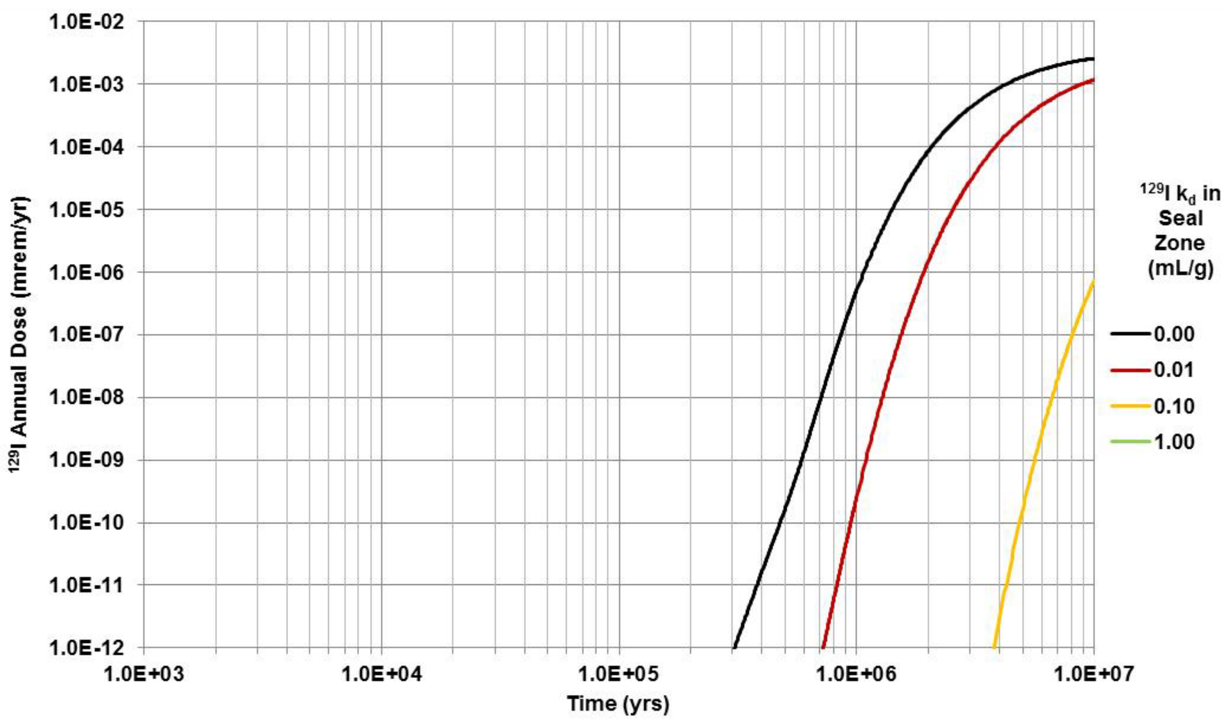


Figure 4-33. Effect of Sorption in the Seal Zone on Annual Dose from ^{129}I in the Deep Borehole GDS Model

Diffusion—The effect of the diffusion coefficient on ^{129}I annual dose is shown in Figure 4-34. The sensitivity analysis includes two cases:

- Baseline Molecular (Free Water) Diffusion Coefficient (2.30×10^{-9} m^2/s)—The corresponding effective diffusion coefficient for ^{129}I in each region is based on the local porosity and tortuosity.
- Enhanced Molecular (Free Water) Diffusion Coefficient (1.15×10^{-8} m^2/s)—This results in a corresponding effective diffusion coefficient for ^{129}I in each region that is a factor of 5 larger than the baseline value. The increased molecular diffusion coefficient reproduces potential changes in the effective diffusion coefficient for ^{129}I that might result from changes in available porosity (e.g., due to anion exclusion) or tortuosity.

The factor-of-5 increase in diffusion coefficient has a significant effect on dose. This is because the corresponding factor-of-5 increase in diffusive flux rate has a significant effect on transport in regions where diffusion is the dominant transport mechanism. Diffusion is dominant at all times in most of the seal zone, which is the slowest transport component in the deep borehole disposal system. Therefore, the increase in diffusion coefficient shifts the dose curve to the left by about a factor of five on the time axis.

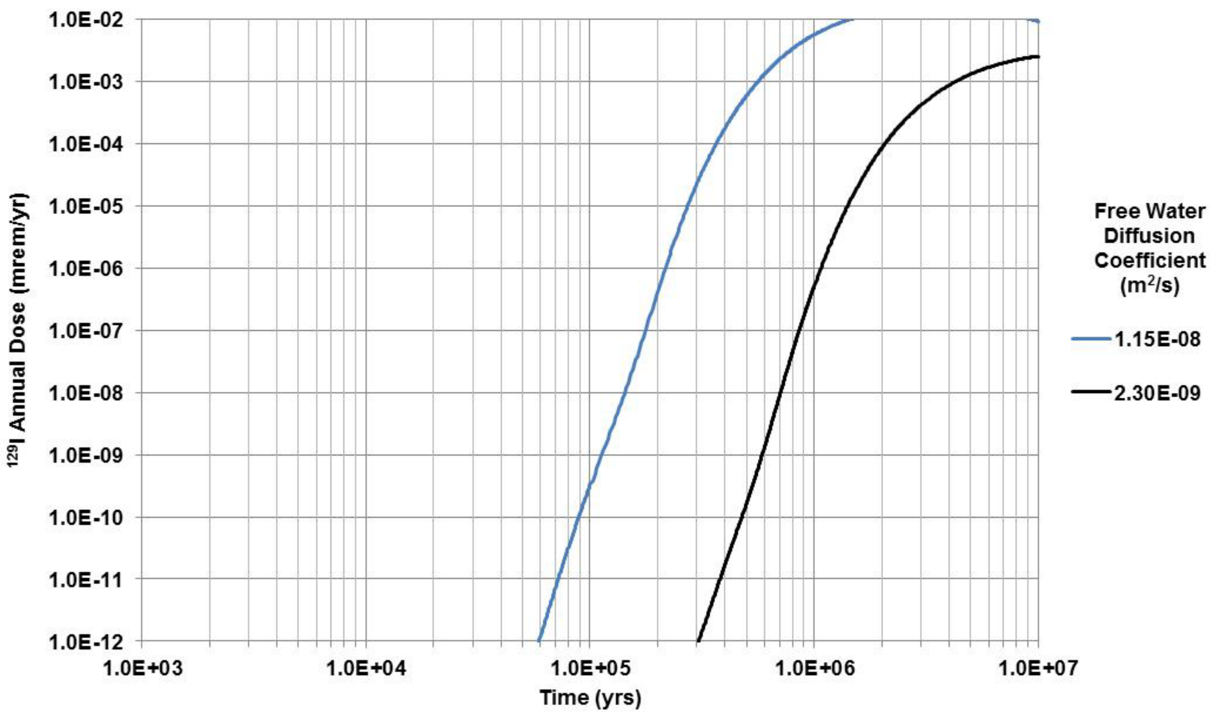


Figure 4-34. Effect of Diffusion Coefficient on Annual Dose from ^{129}I in the Deep Borehole GDS Model

Seal Zone Integrity and Flow Rate—The effects of seal zone integrity were investigated as part of the FY 2011 deep borehole GDS model (Clayton et al. 2011, Section 3.4). To represent degraded and/or defective borehole seals, an upper bound seal zone permeability of 1×10^{-12} m^2 was assumed, as compared to the baseline seal zone permeability of 1×10^{-16} m^2. Probabilistic simulations with the higher seal zone permeability and correspondingly higher flow rates up the borehole produced a mean annual dose (Clayton et al. 2011, Figure 3.4-19) that was several orders of magnitude higher than the baseline dose (Clayton et al. 2011, Figure 3.4-9).

Summary—Based on these four sensitivity analyses examining "one-off" conditions from the baseline scenario, and the effects of seal zone integrity and flow rate investigated in Clayton et al. (2011, Section 3.4), the following observations can be made regarding the performance of a generic deep borehole disposal system under baseline scenario conditions:

- Processes and parameters affecting radionuclide transport through the 1,000-m seal zone can have a significant effect on annual dose. These include sorption, k_d, and seal zone integrity.
- Processes and parameters affecting radionuclide transport through the 2,000-m disposal zone can have a moderate effect on annual dose. Very small increases in sorption, k_d, can noticeably decrease the dose.
- Processes and parameters affecting radionuclide transport through the entire deep borehole disposal system can have a significant effect on annual dose. These include flow rate and diffusion coefficient. These system-wide effects are important in the both disposal zone and the seal zone.
- Processes and parameters affecting waste form degradation can have a moderate effect on annual dose.

5 SYNTHESIS AND CONCLUSIONS: SAFETY ASSURANCES FOR THE FUTURE

5.1 Introduction

While there inevitably will be uncertainty in assessments of safety in the far future for geologic disposal facilities, there are a number of aspects of geologic disposal and the processes that likely will be in place to determine whether or not to develop a geologic disposal facility that provide confidence that safety will be assured. Confidence in the safety of a geologic disposal facility is assessed through quantitative and qualitative aspects. This section briefly describes the methodology for addressing uncertainty in projections of long-term performance, and summarizes the results of deterministic simulations and sensitivity analyses undertaken using the safety assessment models presented in Section 4.4.

The U.S. approach of licensing a geologic disposal facility historically was based on facility development in phases, including authorizations: (i) to construct repositories; (ii) to receive and possess used nuclear fuel and high-level radioactive waste in such repositories; and (iii) for closure and decommissioning of such repositories. This phased approach sets a path for multiple reviews of the technical arguments supporting development of the geologic disposal facility. This process includes the collection of data to confirm that the geologic disposal facility is functioning as planned. This performance confirmation program takes place over the life of the geologic disposal facility, and required data collection, oversight, and monitoring activities provide safety assurances for the future.

The quantitative safety assessments presented in Section 4, supported by the qualitative evidence and justifications presented in Sections 2 and 3, indicate that sufficient information exists at this time to support a conclusion that the U.S. used nuclear fuel and high-level radioactive waste disposal program may continue to progress. There is confidence that identification of regions with potentially acceptable sites, and screening to identify and compare potentially acceptable sites can proceed on the basis of existing information.

This section briefly highlights the intent of Used Fuel Disposition Campaign activities that support continued used nuclear fuel and high-level radioactive waste disposal programs for mined geologic disposal facilities in the salt, clay, and granite, and deep borehole disposal in crystalline rock. The *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) focuses on identifying knowledge gaps and opportunities where research and development have the greatest potential to contribute to advancing the understanding of technical issues regarding the deep geologic disposal of nuclear waste.

An important use of the Generic Safety Case is to inform stakeholders of the progress that has been made relative to the current phase of the program, particularly the issues of concern and the data needed to have confidence to move to the next phase of the program. This section presents a summary of the rationale for concluding that sufficient information exists to support moving forward with the activities needed to develop new legislation, promulgate new safety standards and implementing regulations, and identify regions and potentially acceptable sites.

The Generic Safety Case document is likely to be superseded by one or more site-specific safety case documents. With the safety case containing information about not only the technical information supporting the program, but also the issues that are of interest to the stakeholders, and the current state of how those issues are being addressed, there will be multiple opportunities for input from the stakeholders that can be used by decision makers to guide the direction of the program.

The Used Fuel Disposition Campaign may continue to develop the information in this Generic Safety Case document and, as the program progresses to site-specific investigations, support deliberations of stakeholders and decision makers by moving toward one or more media-specific geologic disposal system safety case documents.

5.2 Confidence in the Face of Uncertainty

5.2.1 Introduction

There is a need to quantify the levels of uncertainty, and understand how that uncertainty propagates through the analyses supporting the safety case for a geologic disposal facility. The information that would be collected through research and development provides a basis to communicate the safety case for a generic geologic disposal system, or a geologic disposal system in a specific medium.

Quantitative safety assessments of the long-term performance of the repository system must consider uncertainty. In accordance with the safety assessment methodology described in Section 2.3.1, uncertainty is incorporated in the safety assessment through (1) identification and screening of potentially relevant FEPs that may have an effect on that performance, (2) construction of scenarios from the screened in FEPs, (3) estimation of the likelihood of occurrence and consequences of the scenarios (and the underlying processes and events), and (4) representation of the conceptual models in the form of numerical models that explicitly capture uncertainty, generally using probabilistic methods. The Used Fuel Disposition Campaign will continue to evaluate ways to reduce uncertainties in repository performance models. Uncertainties will always remain because of the long timeframes over which the system performance must be assessed, the natural variability in features and processes at the site, and limitations on the amount of data that can be collected. The principles of reasonable assurance and reasonable expectation recognize that there will inevitably be uncertainties, and that proof of the future performance of engineered barriers and a geologic setting over time periods of many thousands of years is not to be had in the ordinary sense of the word. Because uncertainty cannot be eliminated, the approach to building confidence in analyses of repository performance relies on multiple lines of evidence. Collectively, these multiple lines of evidence are important to the postclosure safety case.

5.2.2 Uncertainty in System Safety Estimates

As described in Section 2.3.1 (Step 5), three major sources of uncertainty must be considered in problems associated with very long-term projections of geologic disposal system performance: uncertainty in the future state of the system (aleatory uncertainty); uncertainty in the accuracy, appropriateness, and/or completeness of the conceptual and numerical models (model uncertainty); and uncertainty in the data and parameter values due to incomplete knowledge of the present system and the inherent complexity of natural systems (epistemic uncertainty). Thus, there is a need to quantify the levels of uncertainty, and understand how that uncertainty propagates through the analyses supporting the safety case for a geologic disposal system. The information that would be collected through research and development provides a basis to communicate the safety case for a generic repository or geologic disposal system, or a geologic disposal system in a specific medium.

Good management, siting, and engineering principles and practices resulted in all national programs developing disposal strategies that include erring on the side of caution. Under such a principle, siting and design strategies that include developing a robust system are adopted. Robust geologic disposal facilities exhibit an absence of, or relative insensitivity to, detrimental events and processes arising either internally within the disposal system, or externally in the form of geological and climatic phenomena, and uncertainties with the potential to compromise safety. Furthermore, the assessment strategy that is adopted, wherever possible, provides a range of arguments and analyses for the safety case that is well-founded, supported by multiple lines of evidence, and adequately treat uncertainty. The safety case may, for example, take into account all processes that may affect system performance, but in documenting the safety case, emphasis may be placed on a limited number of processes or features relevant to the safety functions of the repository and its environment that are well-understood and reliable, such as long-lived corrosion resistant canisters and stable properties of the host rock. All potentially detrimental processes or features should be disclosed and taken into account in the assessment (Organisation for Economic Co-operation and Development 2004) even if it is possible to screen out their contributions to performance either on the basis of low probability of occurrence or insignificant consequence.

A principal concern for technical analyses of this type might be developing consensus on the input statistical distributions of parameter values for the assessments. Because of the requirements for spatial resolution or the infrequency of particular events, deriving the distributions from measurement programs or from observations might not be feasible for defining the parameter distributions. This means that an element of informed judgment will often be involved. A challenge of capturing and propagating uncertainty is making the probabilistic results transparent and easily understood.

As described in Section 2.3.1 (Step 6), the safety assessments can treat the statistical distributions of parameter values using deterministic or probabilistic methodologies. A deterministic model is one in which the variable states of the model are uniquely determined by parameters in the model and by the previous states of these variables. As a consequence, deterministic models perform the same way for a given set of initial conditions. In a deterministic model, sensitivity can be examined using one-off analyses. A one-off analysis is performed by changing one or more parameter values from the baseline value to other values within its statistical distribution (i.e., within the range of uncertainty) and examining the corresponding effect on system performance. A bounding analysis represents a particular type of deterministic analysis in which the parameter values are chosen such that the performance of the system is at a state of "worst case." It is often difficult to define what the worst case is; however, it is typically easy to defend if all agree that the performance could not be worse than calculated. Conversely, in a probabilistic or stochastic analysis, variable states are not described by unique values, but rather by probability distributions. Consequently, there is not a unique answer represented by the analyses; rather, uncertainty is captured by conducting multiple realizations using values sampled from the input parameter value distributions, and the results are presented as a probabilistic distribution of the outcomes from each realization. A previously agreed upon value of the distribution, such as the mean or median, is defined as the overall "result." Probabilistic analyses, therefore, explicitly account for uncertainty. However, as probabilistic models become more complex (i.e., have more uncertain parameters and/or more interactions between uncertain parameters), it sometimes becomes more difficult to isolate the effects of specific uncertainties.

At the heart of selection of the type of methodology for treating variable data is an interest in realism of the results. If the parameters did not vary in time and spatial dimensions, and if they were known with a high degree of precision, then it would be possible to model system performance deterministically with a "realistic" result. Because the input variables are in fact uncertain, the analyst could turn to the use of bounding values to ensure that the results do not overestimate the performance of the system. To ensure conservative results, pessimistic parameter values are often selected. Analyses using pessimistic scenarios and parameter values can sometimes be more easily understood than probabilistic or stochastic analysis. The results of these conservative calculations are, however, no longer estimates of likely behavior but rather bounding estimates. Bounding estimates can be criticized for compounding conservative assumptions because they can easily produce consequences that are highly improbable. On the other hand, if compliance can be shown with a bounding estimate, then there is no need for a more complex analysis. Bounding estimates can thus be very useful, but care should be given as to how one could combine the robust, bounding-estimate type of assessment with a probabilistic analysis (National Academy of Sciences 1995).

When all reasonable steps have been taken to reduce technical uncertainties by, for example, performing site characterization and material testing programs, there still remains a residual, unquantifiable uncertainty. It can never be totally ruled out that the best analytic conclusions might be affected by some hitherto unknown or overlooked process or event. Additionally, for complex coupled systems, it can be difficult to identify parameter values that are "conservative". The values may be conservative with respect to one aspect or subsystem, but not conservative with respect to the total system. A defense against it is to rely on informed judgment.

Clearly, from a technical perspective, it is possible to address the uncertainties in assessing the performance of nuclear waste repositories. Safety assessments can provide not only the analyses of the future performance of the repository but also estimates of the uncertainty in the safety assessments. The real issue today is whether or not the methods and data available are capable of producing assessments of behavior, including bounding estimates that are technically adequate for their intended use.

An understanding of the performance of the disposal system and its safety features and processes evolves as more data are accumulated and scientific knowledge is developed. Early in the development of the concept, the data and the level of understanding gained should provide the confidence necessary to commit the resources for further investigations. Sensitivity analyses conducted on probabilistic results can be used to identify and prioritize resources to those that most impact disposal system metrics of performance. Before the start of construction, during emplacement and at closure, the level of understanding should be sufficient to support the safety case. It is important to recognize that there are multiple components of uncertainty inherent in modeling complex environmental systems and that there are inevitably significant uncertainties associated with projecting the performance of a geological disposal system.

To support this initial safety case, the effects of uncertainties in the safety assessments for the four baseline scenarios are investigated using one-off deterministic sensitivity analyses. As the safety assessments become more site specific and more information describing parameter values becomes available, probabilistic sensitivity analyses will be performed to provide more detailed insights, particularly into the effects of couplings between uncertain processes and parameters.

5.2.3 Sensitivity Analysis Results

As described in Section 4.3, four generic deterministic safety assessment models were used to give preliminary insight to the possible long-term behavior of the generic geologic disposal facilities. The input parameters were selected from searches of relevant literature and care was used to assure that the values selected were representative of the generic media. With the exception of the assumption of instantaneous waste package failure, no attempt was made to be deliberatively conservative or to bound the results; the baseline deterministic simulation represents a set of parameter values within the distribution of possible values. Because the models were deterministic, no uncertainty quantification was available or attempted in the analysis of results. To investigate the sensitivity of the simple deterministic models to variability in the input parameters, a number of one-off sensitivity studies were performed with each of the models. These sensitivity studies represent a first step in understanding likely variability in predicted safety assessment results. There are several caveats that should be understood in examining these sensitivity analyses, and, for that matter, the results of the safety assessment models themselves. Safety assessment is an iterative process and depends on how the system model is represented. The models represent selected baseline scenarios of undisturbed conditions. While the baseline scenarios are representative of certain generic aspects of the four different disposal systems, they are not necessarily representative of expected or nominal undisturbed conditions. These preliminary iterations of generic deterministic safety assessment models do not have the appropriate pedigree to support compliance determination, e.g. they lack completeness of scenario development, parameter value justification, necessary representations of conceptual models, and representations of site-specific information.

The sensitivity analyses presented in Section 4.4.2 were selected to provide insight into the parameters to which the calculated results of the deterministic safety assessment models were most sensitive. Here too, it is important not to seek too much resolution or coverage in the selection of parameters to vary in the one-off calculations because of the simplified and generic nature of the baseline models. The development of the models provided important insights to determine the appropriate parameters for the sensitivity analyses. The following discussion summarizes the highlights of the results of the deterministic baseline analyses (Section 4.4.1) and the one-off sensitivity analyses (Section 4.4.2) for each of the four generic safety case disposal options. In these analyses, the focus is the impact to the release and migration of ^{129}I, because it is the largest contributor to dose and sometimes the only radionuclide to reach the biosphere. These analyses suggest that it is possible to find a suitable site for a mined geologic disposal facility or for deep borehole disposal.

The deterministic results of the generic salt repository baseline model indicate that radionuclide releases to the receptor location in the biosphere are minimal, and primarily due to long-lived non-sorbing ^{129}I. Radionuclide transport through the engineered barrier system and the near-field salt DRZ between the repository and the underlying interbed is slow due to very low brine flow rates resulting in diffusion-dominated transport. Salt creep closure of the repository excavation and DRZ minimize the potential for high-permeability fracture connections to the underlying interbed. Radionuclide transport through the far-field anhydrite interbed is slow due to very low brine flow rates resulting in diffusion-dominated transport, radionuclide sorption, absence of well-connected fractures in the interbed, and the long migration distance to the receptor location. The sensitivity analyses show that, under baseline salt scenario conditions, the calculation of annual dose is very sensitive to distance to the receptor, brine flow rate, interbed sorption, and the diffusion coefficient. The results are somewhat sensitive to waste form degradation rate, and only slightly sensitive to the near-field salt DRZ integrity.

The deterministic results of the generic clay repository baseline model indicate that radionuclide releases to the receptor location in the biosphere are small. Radionuclide transport through the clay host rock far field is slow due to diffusion-dominated transport, radionuclide sorption, and distance to the receptor (sufficient clay formation thickness). Radionuclide transport through the near field (bentonite buffer and clay DRZ) is slow due to diffusion-dominated transport, radionuclide sorption, and healing in the clay DRZ that minimizes the potential for high-permeability fissure connections to the far-field clay. The sensitivity analyses show that, under baseline clay scenario conditions, the calculation of annual dose is very sensitive to sorption in the far-field clay, flow rate, and distance to the receptor, and somewhat sensitive to waste form degradation rate, and waste package lifetime. Buffer and DRZ integrity, and diffusion coefficient have little effect.

The deterministic results of the generic granite repository baseline model indicate that radionuclide releases to the receptor location in the biosphere are small. Radionuclide transport through the near field (the bentonite buffer and granite DRZ) is slow due to an intact buffer, diffusion-dominated transport in the bentonite, and radionuclide sorption. Radionuclide transport through the far field (granite fracture and matrix) is slow due matrix diffusion associated with fracture transport, radionuclide sorption in the matrix, and long migration distance to the receptor location. The sensitivity analyses show that, under baseline granite scenario conditions, the calculation of annual dose is very sensitive to waste package lifetime, sorption in the far-field granite, distance to receptor, and flow rate through the system, somewhat sensitive to waste form degradation rate and gap fraction, and slightly sensitive to sorption in the bentonite buffer, and fracture spacing. Diffusion coefficient has little effect.

The deterministic results of the generic deep borehole baseline model indicate that radionuclide releases to the receptor location in the biosphere are small. Radionuclide transport through the bentonite seal zone is slow due to very low thermally induced fluid flow rates resulting in diffusion-dominated transport, durability of the seals with respect to hydrological barrier performance with only minor DRZ bypass, radionuclide sorption, and distance to the receptor. Radionuclide transport through the disposal zone is slow due to low thermally induced fluid flow rates that decrease over time, resulting in diffusion-dominated transport after about 700,000 years, radionuclide sorption, and long migration distance (as much as 2,000 m) for the deepest waste packages. Radionuclide transport through the basement deep granite is negligible due to the very low permeability and lack of significant fracture connection to overlying formations. The sensitivity analyses show that, under baseline deep borehole scenario conditions, the calculation of annual dose is very sensitive to sorption in the seal zone and the diffusion coefficient. The results are somewhat sensitive to the waste form degradation rate, and sorption in the disposal zone. In addition, based on previous simulations (Clayton et al. 2011, Section 3.4), deep borehole system performance is sensitive to seal zone integrity and flow rate up the borehole.

5.3 Future Activities

5.3.1 Used Fuel Disposition Research and Development Activities

The *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) has been developed to document Used Fuel Disposition Campaign research needs and priorities. The prioritization information will be maintained and updated as:

- The U.S. program evolves,
- The UFD FEPs, which provide the basic organizational structure for the Roadmap, are revised,
- Research and development topics are identified and subsequently mapped to issues within the Roadmap, and
- Research and development activities are completed necessitating an update to the information and reprioritization of the issues.

The evolving *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) will help to ensure that the technical information needed for managing the back end of the nuclear fuel cycle is available when needed. The Roadmap is focused on identifying knowledge gaps and opportunities where research and development has the greatest potential to contribute to advancing the understanding of technical issues regarding the deep geologic disposal of nuclear waste. Initial research and development opportunities were identified, for each of the four disposal options, in terms of the UFD FEP(s) they were most likely to impact.

Table 5-1 reproduces the results of the priority ranking for the natural system. Shading for an entry indicates that research in that area has been undertaken in other repository programs. The highest ranked parameters are flow and transport pathways in granite/crystalline repositories, the DRZ for deep borehole disposal and clay/shale repositories, hydrologic processes for salt repositories, chemical processes for clay/shale repositories, and thermal processes for clay/shale repositories. To compare this to the results of the safety assessments and sensitivity analyses presented in Section 4.4, it first must be remembered that those analyses are based on deterministic evaluations of baseline undisturbed scenarios; not all nominal processes were accounted for, and disruptive events, such as long-term tectonic processes, seismic activity, and climatic processes were not explicitly accounted for because of their reliance on site specific information.

The sensitivity analyses for granite disposal show that the results are very sensitive to flow rate and sorption (k_d's), and somewhat sensitive to waste form degradation rate and gap fraction. The sensitivity analyses for deep borehole disposal show that the results are very sensitive to sorption in the seal zone. The sensitivity analyses for salt disposal show that the results are very sensitive to flow rate, and the radionuclide diffusion coefficient for ^{129}I. The sensitivity analyses for clay disposal show that the results are very sensitive to flow rate and sorption (k_d's). Although these sensitivity analyses are based on models that did not necessarily account for all nominal processes, these results are still reasonably consistent with the Roadmap results of the priority ranking for the natural system. As Used Fuel Disposition Campaign research activities are prioritized based on this ranking matrix, and the results of the safety assessments presented in Section 4.4 are consistent with the matrix, there can be confidence that research activities are appropriately focused at this time.

Table 5-1. Synopsis of the Results of the Priority Ranking for the Natural System

GEOSPHERE	Granite / Crystalline	Deep Borehole	Salt	Clay / Shale
1.2.01. LONG-TERM PROCESSES (tectonic activity)	Low	Low	Low	Low
1.2.03. SEISMIC ACTIVITY - Effects on EBS - Effects on Natural System	 High Low	 High Low	 High Low	 High Low
1.3.01. CLIMATIC PROCESSES AND EFFECTS	Low	Low	Low	Low
2.2.01. EXCAVATION DISTURBED ZONE (EDZ)	Medium	High	Medium	High
2.2.02 HOST ROCK (properties)	High	High	High	High
2.2.03 OTHER GEOLOGIC UNITS (properties)	Medium	Medium	Medium	Medium
2.2.05. FLOW AND TRANSPORT PATHWAYS	Medium	Medium	Low	Medium
2.2.07. MECHANICAL PROCESSES	Low	Low	Medium	Medium
2.2.08. HYDROLOGIC PROCESSES	Low	Medium	High	Medium
2.2.09. CHEMICAL PROCESSES - CHEMISTRY	Low	Medium - High	Low – Medium	Medium - High
2.2.09. CHEMICAL PROCESSES - TRANSPORT	Medium	Medium - High	Medium – High	Medium
2.2.10. BIOLOGICAL PROCESSES	Low	Low	Low	Low
2.2.11. THERMAL PROCESSES	Low	Medium	Low	Medium
2.2.12. GAS SOURCES AND EFFECTS	Low	Low	Low	Low
2.2.14. NUCLEAR CRITICALITY	Low	Low	Low	Low

NOTE: NS = natural system
Shading for an entry indicates that research in that area has been undertaken in other repository programs.

Source: U.S. Department of Energy 2012, Table 7.

Similarly, Table 5-2 reproduces the results of the priority ranking for the engineered system, including the waste form and waste package, buffer materials and seal and other materials. Shading for an entry indicates that research in that area has been undertaken in other repository programs. The presentation is broken down by engineered component and likely materials that could be used. For the inventory and waste form, issues related to ceramics and metals ranked higher than those for used nuclear fuel and high-level radioactive waste glass. For the waste package materials, issues related to novel materials generally ranked higher than those for steel and copper. In the categories of engineered system materials, for buffer and backfill materials, issues related to mixed material backfills and buffers generally ranked higher than those for clay, salt, or crystalline buffer and backfill materials. For seal and liner materials, issues related to polymer materials generally ranked higher than those for cement, asphalt, or metal seal and liner materials. For other engineered barrier materials, issues related to barrier additive materials ranked slightly higher than issues related to low pH cements, salt-saturated cements, or geopolymers. To compare this to the results of the safety assessments and sensitivity analyses presented in Section 4.4, it first must be remembered that those analyses are based on deterministic evaluations of the baseline undisturbed scenarios; not all nominal processes were accounted for, and disruptive events, such as long-term tectonic processes, seismic activity, and climatic processes were not explicitly considered.

The sensitivity analyses for granite disposal show that the results are very sensitive to waste package lifetime and somewhat sensitive to waste form degradation rate and gap fraction. The sensitivity analyses for deep borehole disposal show that the results are somewhat sensitive to the waste form degradation rate. The sensitivity analyses for salt disposal show that the results are somewhat sensitive to waste form degradation rate. The sensitivity analyses for clay disposal show that the results are somewhat sensitive to waste form degradation rate and waste package lifetime. These limited results are not inconsistent with the Roadmap results of the priority ranking for the engineered system: waste form and waste package; however, it is recommended that more detailed sensitivity analyses be conducted to further evaluate the Roadmap prioritization.

The *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) will be used in support of, and in conjunction with, the Generic Safety Case to identify and provide the best generic data available to plan future research and development activities. This will include developing the justifications in the Roadmap to illustrate priorities appropriate to support future actions, such as screening activities.

A principal use of research and development related to a nuclear waste geologic disposal facility is in the area of confidence building. Research programs can be selected to support broad confidence building and education efforts with stakeholders and the public. Geologic disposal program development will benefit from the information developed for the generic geologic disposal system investigations. The qualitative and quantitative discussions presented in this document can be used by stakeholders and policymakers to better understand the evolving science and implications thereof.

Table 5-2. Synopsis of the Results of the Priority Ranking for the Engineered System

WASTE MATERIALS → UNF, Glass, Ceramic, Metal	
2.1.01.01, .03, .04: INVENTORY	Low
2.1.02.01, .06, .03, .05: WASTE FORM	High
WASTE PACKAGE MATERIALS → Steel, Copper, Other Alloys, Novel* Materials	
2.1.03.01, .02, .03, .04, .05, .08: WASTE CONTAINER	High
2.1.07.03, .05, .06, .09: MECHANICAL PROCESSES	Medium
2.1.08.02, .07, .08: HYDROLOGIC PROCESSES	Low
2.1.09.01, .02, .09, .13: CHEMICAL PROCESSES – CHEMISTRY - Radionuclide speciation/solubility	Medium High
2.1.09.51, .52, .53, .54, .55, .56, .57, .58, .59: CHEMICAL PROCESSES - TRANSPORT - Advection, diffusion, and sorption	Low Medium
2.1.10.x: BIOLOGICAL PROCESSES (no FEPs were scored in this category)	Low
2.1.11.01, .02, .04: THERMAL PROCESSES	Medium
2.1.12.01: GAS SOURCES AND EFFECTS	Low
2.1.13.02: RADIATION EFFECTS	Low
2.1.14.01: NUCLEAR CRITICALITY	Low
BUFFER / BACKFILL MATERIALS → Cementitious, Bituminous, Mixed Materials: clay, salt, crystalline environments	
2.1.04.01: BUFFER/BACKFILL	High
2.1.07.02, .03, .04, 09: MECHANICAL PROCESSES	Medium
2.1.08.03, .07, .08: HYDROLOGIC PROCESSES	Medium
2.1.09.01, .03, .07, .09, .13: CHEMICAL PROCESSES – CHEMISTRY - Radionuclide speciation/solubility	Medium High
2.1.09.51, .52, .53, .54, .55, .56, .57, .58, .59, .61: CHEMICAL PROCESSES – TRANSPORT - Colloid facilitated transport	Medium Low
2.1.10.x: BIOLOGICAL PROCESSES (no FEPs were scored in this category)	Low
2.1.11.04: THERMAL PROCESSES	Medium
2.1.12.01, .02, .03: GAS SOURCES AND EFFECTS	Medium
2.1.13.02: RADIATION EFFECTS	Low
2.1.14.02: NUCLEAR CRITICALITY	Low

Table 5-2. Synopsis of the Results of the Priority Ranking for the Engineered System (continued)

SEAL / LINER MATERIALS → Cementitious, Asphalt, Metal, Polymers	
2.1.05.01: SEALS	Medium
2.1.06.01: OTHER EBS MATERIALS	Medium
2.1.07.02, .08,.09: MECHANICAL PROCESSES	Medium
2.1.08.04, .05, .07, .08, .09: HYDROLOGIC PROCESSES - Flow through seals	Low Medium
2.1.09.01, .04, .07, .09, .13: CHEMICAL PROCESSES CHEMISTRY - Radionuclide speciation/solubility	Medium High
2.1.09.51, .52, .53, .54, .55, .56, .57, .58, .59: CHEMICAL PROCESSES – TRANSPORT - Advection, diffusion, and sorption	Low Medium
2.1.10.x: BIOLOGICAL PROCESSES (no FEPs were scored in this category)	Low
2.1.11.04: THERMAL PROCESSES	Medium
2.1.12.02, .03: GAS SOURCES AND EFFECTS	Low
2.1.13.02: RADIATION EFFECTS	Low
2.1.14.02: NUCLEAR CRITICALITY	Low
OTHER MATERIALS → Low pH Cements, Salt-Saturated Cements, Geo-polymers, Barrier Additives	
2.1.06.01: OTHER EBS MATERIALS	Medium
2.1.07.08, .09: MECHANICAL PROCESSES	Medium
2.1.08.04, .05: HYDROLOGIC PROCESSES	Medium
2.1.09.04, .07, .09, .13: CHEMICAL PROCESSES - CHEMISTRY - Radionuclide speciation/solubility	Medium High
2.1.09.51, .52, .53, .54, .55, .56, .57, .58, .59: CHEMICAL PROCESSES – TRANSPORT - Advection, diffusion, and sorption	Low Medium
2.1.10.x: BIOLOGICAL PROCESSES (no FEPs were scored in this category)	Low
2.1.11.04 THERMAL PROCESSES	Medium
2.1.12.02, .03: GAS SOURCES AND EFFECTS	Low
2.1.13.02: RADIATION EFFECTS	Low
2.1.14.02: NUCLEAR CRITICALITY	Low

Notes: * A novel engineered barrier system material refers either to a new material designed for improved performance within a geologic disposal system or an existing material that has not been extensively studied and used in the design of a geologic disposal system that could lead to improved performance.

1. Shading for an entry indicates that research in that area has been undertaken in other geologic disposal programs.
2. FEP number lists delimited by commas show only the change in the fourth field of the FEP.

Source: U.S. Department of Energy 2012, Table 8

5.3.2 Additional Data Will Be Collected To Support the Siting Evaluations

The process of siting and development of a geologic disposal facility involves the continuing pursuit of new information, driven by issues that have been identified as development of the site matures, and the continual reevaluation of the meaning of that new information. The role of the safety case documents in this process is to collect information about the issues that are of interest to the stakeholders, regulators, and decision makers, and evaluate the state of resolution of those issues in light of new data that has been collected as the program proceeds. The site characterization process will continue until there is enough information to justify a recommendation to move forward with a site for licensing.

There will be multiple opportunities to evaluate new data collected at each step of the process of siting and development of the geologic disposal facility. As has been noted, it is envisioned that the Generic Safety Case will be revised and mature to site-specific safety case documents, and be able to provide information at each of these stages to facilitate discussion of issues associated with geologic disposal system development. To proceed from one of the steps to the next will require the collection of significant amounts of data and analysis of that data. These data will not only allow issues that have been identified to be resolved, there is also likelihood that the new information collected will identify the need to revisit how the performance of the system is measured.

The process involves the continuing pursuit of new information, driven by issues that have been identified as development of the site matures, and the continual reevaluation of the meaning of that new information. The role of the safety case documents in this process is to collect information about the issues that are of interest to the stakeholders, regulators, and decision makers, and evaluate the state of resolution of those issues in light of new data that have been collected as the characterization of the sites proceeds. Under a safety case document approach, issues raised by the regulator and any oversight agency can be captured and treated equally in terms of reaching the ultimate goal of the safety case, that is, to assist in the resolution and documentation of the important issues of all stakeholders, regulators, and policy and decision makers. A regularly updated safety case document, used by all potential parties to the development of a geologic disposal system, could identify and track issues formally.

5.3.3 Oversight Increases If Construction Authorization Is Granted

Federal law makes a geologic disposal system subject to regulation by the U.S. Nuclear Regulatory Commission. The U.S. Nuclear Regulatory Commission will review every aspect of the project, which includes evaluating scientific work and system performance. It is expected that there will also be oversight by the U.S. Nuclear Waste Technical Review Board, and states and affected units of local government. Oversight becomes more formal, and increases, as the program progresses.

A performance confirmation program would be conducted to evaluate the adequacy of assumptions, data, and analyses that led to the findings that permitted construction of the geologic disposal facility and subsequent emplacement of the wastes. A performance confirmation program would begin during site characterization. The performance confirmation program can be informed using sensitivity analyses from the safety assessment particularly in identifying what should be monitored and within what limits the monitored metric is acceptable. The monitoring program would involve activities that provide an increased understanding of the processes that are important to repository safety. These activities include testing the repository environment (e.g., rock properties, chemistry) and verifying the data with the results predicted by computer models. Key geotechnical and design parameters, including any interactions between natural and engineered systems and components, would be monitored throughout site characterization, construction, emplacement, and operation to identify any significant changes in the disposal system conditions. The data generated from performance confirmation would provide additional confidence in the results of the safety assessments.

The geologic disposal facility operations area must be designed to preserve the option of waste retrieval for a specified period. The performance confirmation program will provide data that indicates, where practicable, whether (1) the actual subsurface conditions encountered and changes in those conditions during construction and waste emplacement operations are within the design limits, and (2) whether the natural and engineered systems and components required for geologic disposal facility operation, or which are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated.

5.3.4 Regulatory Requirements Will Assure Monitoring and Physical Protection into the Distant Future

A phased approach to repository development and licensing implies phased regulatory review processes, examination of facility development, and plans for future to ensure safety and health, including examination of plans for the post-closure monitoring of the geologic repository.

As described in Section 5.3.3, the performance confirmation program would begin with site characterization and continue through the construction, emplacement, and operational phases of the geologic disposal facility. An important aspect of this program and the license itself is the collection and preservation of records. As it is possible that decisions could be made decades or more into the future, a robust records program is needed to ensure that the data upon which decisions were made is available and reviewable. As a result, before the facility could be closed, all of the important provisions on which the license was predicated would be reviewed using data and analyses collected over the life of the facility. With an appropriate repository design, it could be possible to allow future generations the option to decide whether to close the repository or continue monitoring it.

Plans may also be developed for a program for continued oversight to prevent any activity at the site that poses an unreasonable risk of breaching the geologic repository's engineered barriers or increasing the exposure of individual members of the public to radiation beyond allowable limits. Thus, while the closure of the repository is intended to be a final act, continued monitoring of the facility can ensure the protection of those that could potentially be affected by the facility.

Requirements that are being developed internationally (El Baradei 2003) to deal with the monitoring and protection of closed repositories may provide guidance:

> *Geological repositories, after closure, are expected to achieve adequate long term safety without the need for reliance on continuing institutional controls. However, the need to meet IAEA safeguards requirements is likely to result in some long term monitoring and possibly other forms of institutional controls for disposal facilities, particularly those that contain spent fuel subject to safeguards. These controls must be sufficiently robust to address non-proliferation and security concerns, in a manner that enhances public confidence — and they must be adequate to ensure stability well into the future. The IAEA is currently developing site-specific safeguards requirements and long term surveillance and monitoring approaches.*

5.4 Conclusion: The Potential Postclosure Performance of the Generic Disposal Systems Has Been Shown to be Likely to be Safe

The evaluations presented in this Generic Safety Case demonstrate that, based on existing information, reasonable designs, and generic safety assessment models, it is likely that mined geologic disposal systems in salt, clay, and granite, or deep borehole disposal facilities in crystalline rock can be developed in the U.S. that are capable of safely isolating used nuclear fuel and high-level radioactive waste from the public and the environment. While the models used in the initial safety assessments presented in this Generic Safety Case are preliminary, simplified, and scenario-specific, efforts have been made to ensure that the analyses are cautious and do not overestimate performance. The analyses include sensitivity studies to examine the range of performance considering potential uncertainties in the input parameter values. Qualitative information on the natural and engineered barriers systems, including studies of natural analogues has been examined as well to support the conclusion that the geologic disposal facility will perform safely for very long times. The preponderance of the evidence obtained and analyzed suggests that there is a basis for having sufficient confidence in the system's long-term safety now to allow moving forward.

In preparing the *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012), the experts that participated in the elicitations also concluded that sufficient information existed at this time to undertake site screening. This is not an unexpected conclusion; a considerable amount of generic and non-site-specific data now exists for the four geologic disposal options that are currently being examined, including information gained from domestic and foreign geologic disposal studies. The plans developed under the *Used Fuel Disposition Campaign Research and Development Roadmap* (U.S. Department of Energy 2012) are designed to move quickly to focused research for specific media, and then specific regions and sites, as appropriate. Studies will continue as decisions are made and sites selected for characterization. The process of data collection, issue identification, and issue resolution will continue in parallel with the evolution of the site specific designs and safety assessment models. This will be repeated through each phase of the repository development program. Safety will be thoroughly investigated before the final closure of the repository. Monitoring will continue into the future. Over time, there is every reason to expect there will be an increasingly more thorough understanding of the degree of long-term safety provided by the geologic disposal facility, and its scientific basis. There is also an expectation for increasing confidence in safety over time.

6 REFERENCES

40 CFR Part 141. *National Primary Drinking Water Standards.* Readily available.

40 CFR Part 191. *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*. Readily available.

40 CFR Part 197. *Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada*. Readily available.

Anderson, V.K. 2004. *An Evaluation of the Feasibility of Disposal of Nuclear Waste in Very Deep Boreholes.* Cambridge, MA: Massachusetts Institute of Technology Department of Nuclear Engineering.

Andra (National Radioactive Waste Management Agency). 2005a. *Dossier 2005, Andra Research on the Geological Disposal of High-level Long-lived Radioactive Waste: Results and Perspectives*. Paris, France: Andra. www.andra.fr

Andra. 2005b. *Dossier 2005 Argile SYNTHESIS: Evaluation of the Feasibility of a Geological Repository in an Argillaceous Formation, Meuse/Haute-Marne Site.* 2nd Edition. December 2005. Paris, France: Andra. www.andra.fr

Andra. 2005c. *Dossier 2005 Argile Tome: Architecture and Management of a Geological Repository, Meuse/Haute-Marne Site.* December 2005. Paris, France: Andra. www.andra.fr

Andra. 2005d. *Dossier 2005 Argile Tome: Safety Evaluation of a Geological Repository*. Paris, France: Andra. www.andra.fr

Andra. 2005e. *Dossier 2005 Granite: Safety Analysis of a Geological Repository*. Paris, France: Andra. www.andra.fr

Arnold, B.W., P.V. Brady, S.J. Bauer, C. Herrick, S. Pye, and J. Finger. 2011. *Reference Design and Operations for Deep Borehole Disposal of High-Level Radioactive Waste*. SAND2011-6749. Albuquerque, NM: Sandia National Laboratories.

Arnould, M. 2006. "Discontinuity networks in mudstones: a geological approach. Implications for radioactive wastes isolation in deep geological formations in Belgium, France, Switzerland." *Bulletin of Engineering Geology and the Environment* 65:413–422.

Atomic Energy Act. 1954. *Atomic Energy Act of 1954.* Public Law 83-703. August 30, 1954.

Bailey, L., D. Becker, T. Beuth, M. Capouet, J.L. Cormenzana, M. Cuñado, D.A. Galson, L. Griffault, J. Marivoet, and C. Serres. 2011. *PAMINA (Performance Assessment Methodologies in Application to Guide the Development of the Safety Case): European Handbook of the State-of-the-Art of Safety Assessments of Geological Repositories—Part 1.* Deliverable (D-N°: 1.1.4). European Commission. http://www.ip-pamina.eu/

Banwart, S.A., E. Gustafsson, and M. Laaksharju. 1999. "Hydrological and Reactive Processes During Rapid Recharge to Fracture Zones - The Aspo Large Scale Redox Experiment." *Applied Geochemistry* 14(7):873-892.

Bates, R.L. and J.A. Jackson, eds. 1980. *Glossary of Geology.* 2nd Ed. Falls Church, VA: American Geological Institute.

Beauheim, R.L. and R.M. Roberts. 2002. "Hydrology and Hydraulic Properties of a Bedded Evaporite Formation." *Journal of Hydrology* 259(1–4): 66–88.

Bennett, D.G. 2010. *The Joint EC/NEA Engineered Barrier System Project: Synthesis Report (EBSSYN), Final Report, Seventh Framework Programme*. EUR 24232. European Commission.

Bennett, D.G., A.J. Hooper, S. Voinis, and H. Umeki. 2006. The Role of the Engineered Barrier System in Safety Cases for Geological Radioactive Waste Repositories: An NEA initiative in Co-operation with the EC. *Scientific Basis for Nuclear Waste Management XXIX, Materials Research Society Symposium Proceedings* **932**: 43-52.

Bernier, F., X.L. Li, W. Bastiaens, L. Ortiz, M. Van Geet, L. Wouters, B. Frieg, P. Blümling, J. Desrues, G. Viaggiani, C. Coll, S. Chanchole, V. De Greef, R. Hamza, L. Malinsky, A. Vervoort, Y. Vanbrabant, B. Debecker, J. Verstraelen, A. Govaerts, M. Wevers, V. Labiouse, S. Escoffier, J.-F. Mathier, L. Gastaldo, and Ch. Bühler. 2007. *Fractures and self-healing within the excavation disturbed zone in clays (SELFRAC)*. Report EUR 22585. European Commission.

Bethke, C.M. and P.V. Brady. 2000. How the K_d Approach Undermines Groundwater Cleanup. *Groundwater* 38: 435-443.

Blue Ribbon Commission on America's Nuclear Future. 2012. *Report to the Secretary of Energy*.

Blümling, P., F. Bernier, P. Lebon, and C.D. Martin. 2007. The Excavation Damaged Zone in Clay Formations Time-Dependent Behaviour and Influence on Performance Assessment. *Physics and Chemistry of the Earth* **32**(8-14):588-599.

BMWi (Federal Ministry of Economics and Technology). 2008. *Final disposal of high-level radioactive waste in Germany—The Gorleben Project*. Berlin, Germany: Federal Ministry of Economics and Technology. Public Relations/IA8. www.bmwi.de

Bonano, E.J., D. Kessel, M. Marietta, P. Swift and L. Dotson. 2010. *Performance Assessment for Radioactive Waste Management at Sandia National Laboratories: A 30-year History*. SAND2010-0807C. Sandia National Laboratories.

Bradbury, M.H. and B. Baeyens. 2003. “Porewater Chemistry in Compacted Re-saturated MX-80 Bentonite.” *Journal of Contaminant Hydrology* 61:329-338.

Brady, P.V., B.W. Arnold, G.A. Freeze, P.N. Swift, S.J Bauer, J.L. Kanney, R.P. Rechard, and J.S. Stein. 2009. *Deep Borehole Disposal of High-Level Radioactive Waste*. SAND2009-4401. Albuquerque, NM: Sandia National Laboratories.

Bredehoeft, J.D., A.W. England, D.B. Steward, N.J. Trask, and I.J. Winograd. 1978., Geological Disposal of High-Level Radioactive Waste: Earth-Science Perspectives. *Circular 779*. U.S. Geological Survey.

Bruno, J. and E. Cera. 2006. “Spent Fuel” in: (R. Ewing and A. Macfarlane, eds.) *Uncertainty Underground: Yucca Mountain and the Nation’s High-Level Nuclear Waste*. Cambridge, MA: MIT Press.

BSC (Bechtel-SAIC Company). 2004a. *Defense HLW Glass Degradation Model*. ANL-EBS-MD-000016 REV 02, DOC.20041020.0015. Las Vegas, NV: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

BSC. 2004b. *Natural Analogue Synthesis Report*. TDR-NBS-GS-000027 REV 01, DOC.20040524.0008. Las Vegas, NV: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

BSC. 2005a. *Cladding Degradation Summary for LA*. ANL-WIS-MD-000021 REV 03, DOC.20070614.0002. Las Vegas, NV: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

BSC. 2005b. *The Development of the Total System Performance Assessment-License Application Features, Events, and Processes*. TDR-WIS-MD-000003 REV 02, ACC: DOC.20050829.0004. Las Vegas, NV: Bechtel SAIC Company.

Carbol P. and I. Engkvist. 1997. *Compilation of Radionuclide Sorption Coefficients for Performance Assessment*. SKB R-97-13, Svensk Kärnbränslehantering AB.

Carter, J.T. and A.J. Luptak. 2010. *Fuel Cycle Potential Waste Inventory for Disposition*. FCR&D-USED-2010-000031, Rev 2. Washington, DC: U.S. Department of Energy, Office of Used Fuel Disposition.

Carter, J.T., F. Hansen, R. Kehrman, and T. Hayes. 2011a. *A Generic Salt Repository for Disposal of Waste from a Spent Nuclear Fuel Recycle Facility*. SRNL-RP-2011-00149, Rev. 0. Aiken, SC: Savannah River National Laboratory.

Carter, J.T., R.H. Jones Jr., and A.J. Luptak. 2011b. *U.S. Radioactive Waste Inventory and Characteristics Related to Potential Future Nuclear Energy Systems*. FCRD-USED-2011-000068, Rev 2.

Carter, J.T., A.J. Luptak, and J. Gastelum. 2011c. *Fuel Cycle Potential Waste for Disposition*. FCR&D-2010-00031, Rev 3.

Carter, L.J. 1987. *Nuclear Imperatives and Public Trust: Dealing with Radioactive Waste.* Washington, DC: Resources for the Future.

Clayton, D., G. Freeze, T. Hadgu, E. Hardin, J. Lee, J. Prouty, R. Rogers, W.M. Nutt, J. Birkholzer, H.H. Liu, L. Zheng, and S. Chu. 2011. *Generic Disposal System Modeling—Fiscal Year 2011 Progress Report.* FCRD-USED-2011-000184, SAND2011-5828P. Albuquerque, NM: Sandia National Laboratories.

Cranwell, R.M., R.V. Guzowski, J.E. Campbell, and N.R. Ortiz. 1990. *Risk Methodology for Geologic Disposal of Radioactive Waste: Scenario Selection Procedure*. NUREG/CR-1667; SAND80-1429. Albuquerque, NM: Sandia National Laboratories.

Croff, A.G., T.F. Lomenick, R.S. Lowrie, and S.H. Stow. 2003. *Evaluation of Five Sedimentary Rocks Other than Salt for Geologic Repository Siting Purposes, Volume 1: Main Report*. ORNL/TM-2003/256/V1. Oak Ridge, TN: Oak Ridge National Laboratory. [Note: While the work described in the report was completed in the 1984 – 1986 timeframe, U.S. Department of Energy direction to cease work on the report caused these results not to be published. Subsequently, the Department directed that this report be published as it existed at the time work ceased.]

Delage, P., Y.J. Cui, and A.M. Tang. 2010. “Clays in radioactive waste disposal.” *Journal of Rock Mechanics and Geotechnical Engineering* **2**(2): 111–123. Journal online: www.rockgeotech.org

Ekren, E.B., G.A. Dinwiddie, J.W. Mytton, W. Thordarson, J.E. Weir, E.N. Hinrichs, and L.J. Schroder. 1974. *Geologic and Hydrologic Considerations for Various Disposal Concepts of High-Level Radioactive Waste Disposal in Conterminous United States*. Open-File Report 74-158. Reston, VA: U.S. Geological Survey.

El Baradei, M. 2003. *Geological Repositories: The Last Nuclear Frontier*. International Conference on Geological Repositories: Political and Technical Progress, 8-10 December 2003. Stockholm, Sweden.

EPRI (Electric Power Research Institute). 2010. *EPRI Review of Geologic Disposal for Used Fuel and High Level Radioactive Waste, Volume III—Review of National Repository Programs*. Palo Alto, CA: EPRI.

EURATOM. 2011. *Establishing a Community Framework for the Responsible and Safe Management of Spent Fuel and Radioactive Waste.* Council Directive 2011/70/ of 19 July 2011. *Official Journal of the European Union*.

Freeze, G.A., N.S. Brodsky, and P.N. Swift. 2001. *The Development of Information Catalogued in REV00 of the YMP FEP Database*. TDR-WIS-MD-000003 REV 00 ICN 01. Las Vegas, NV: Bechtel SAIC Company.

Freeze, G. P. Mariner, J.E. Houseworth, and J.C. Cunnane. 2010. *Used Fuel Disposition Campaign Features, Events, and Processes (FEPs): FY10 Progress Report*. SAND2010-5902. Albuquerque, NM: Sandia National Laboratories.

Freeze, G., P. Mariner, J.A. Blink, F.A. Caporuscio, J.E. Houseworth, and J.C. Cunnane. 2011. *Disposal System Features, Events, and Processes (FEPs) Progress Report.* SAND2011-6059P. Albuquerque, NM: Sandia National Laboratories.

Freeze, G., C.D Leigh, S.D. Sevougian, and M. Gross. 2012. *A Safety Framework for Disposal of Heat-Generating Waste in Salt: Annotated Outline*. SAND2012-10797P. Sandia National Laboratories. FCRD-UFD-2012-000431.

Freeze, R.A. and J.A. Cherry. 1979. *Groundwater*. Englewood Cliffs, NJ: Prentice-Hall.

Garisto, F., T. Kempe, and P. Gierszewski. 2009. *Technical Summary of the Safety Aspects of the Deep Geological Repository Concept for Used Nuclear Fuel*. NWMO TR-2009-12. Toronto, Ontario: Nuclear Waste Management Organization.

Garrick, B.J. 2003. “The Current Status, Safety, and Transportation of Spent Nuclear Fuel.” *The Bridge: Linking Engineering and Society* (Fall 2003). Washington, DC: U.S. National Academy of Engineering. 11-17.

Gascoyne, M., C.C. Davison, J.D. Ross, and R. Pearson. 1987. *Saline Groundwaters and Brines in Plutons in the Canadian Shield." Saline Water and Gases in Crystalline Rocks*. Special Paper 33:53-68. P. Fritz and S.K. Frape (eds.). Waterloo, Ontario: Geological Association of Canada.

Gierszewski, P., J. Avis, N. Calder, A. D'Andrea, F. Garisto, C. Kitson, T. Melnyk, K. Wei, and L. Wojciechowski. 2004. *Third Case Study - Postclosure Safety Assessment*. Report No: 06819-REP-01200-10109-R00. Toronto, Ontario: Ontario Power Generation.

Goff, K.M. and M.F. Simpson. 2009. Dry Processing of Used Nuclear Fuel. Paper 9516. *Proceedings of Global 2009*, Paris, France (September 6-11, 2009).

Goldsim Technology Group. 2010a. *GoldSim Probabilistic Simulation Environment, User's Guide. Version 10.5*. Issaquah, WA: GoldSim Technology Group LLC.

Goldsim Technology Group. 2010b. *GoldSim Contaminant Transport Module, User's Guide: Version 10.5.* Issaquah, WA: GoldSim Technology Group LLC.

Gonzales, S. and K.S. Johnson. 1984. *Shale and other argillaceous strata in the United States*. ORNL/Sub/84-64794/1. Oak Ridge, TN: Oak Ridge National Laboratory.

Goodwin, B.W., M.E. Stephens, C.C. Davison, L.H. Johnson, and R. Zach. 1994. *Scenario Analysis for the Postclosure Assessment of the Canadian Concept for Nuclear Fuel Waste Disposal*. AECL-10969. Pinawa, Manitoba, Canada: Atomic Energy of Canada Limited (AECL), Whiteshell Laboratories.

Hansen, F.D. and T.J. Vogt. 1987. *Thermomechanical properties of selected shales*. ORNL/Sub/85-97343/2. Oak Ridge, TN: Oak Ridge National Laboratory.

Hansen, F.D. and C.D. Leigh. 2011. *Salt Disposal of Heat-Generating Nuclear Waste.* SAND2011-0161. Albuquerque, NM: Sandia National Laboratories.

Hansen, F.D., E.L. Hardin, R.P. Rechard, G.A. Freeze, D.C. Sassani, P.V. Brady, C.M. Stone, M.J. Martinez, J.F. Holland, T. Dewers, K.N. Gaither, S.R. Sobolik, and R.T. Cygan. 2010. *Shale Disposal of U.S. High-Level Radioactive Waste*. SAND2010-2843. Albuquerque, NM: Sandia National Laboratories.

Hansen, F.D., E. Hardin, and A. Orrell. 2011. Geologic Disposal Options in the USA. SAND2010-7975C. *Waste Management (WM) Conference 2011 Proceedings.* Phoenix, AZ (February 27-March 3, 2011).

Hardin, E., J. Blink, H. Greenberg, M. Sutton, M. Fratoni, J. Carter, M. Dupont and R. Howard. 2011a. *Generic Repository Design Concepts and Thermal Analysis (FY11)*. FCRD-USED-2011-000143, Rev. 2. U.S. Department of Energy, Office of Used Fuel Disposition.

Hardin, E., F.D. Hansen, and S.A. Orrell. 2011b. *Geologic Disposal Options in the USA.* SAND2011-0867C. Las Vegas, NV: U.S. Nuclear Waste Technical Review Board Winter Meeting.

Hardin, E., T. Hadgu, D. Clayton, R. Howard, H. Greenberg, J. Blink, M. Sharma, M. Sutton, J. Carter, M. Dupont, and P. Rodwell. 2012. *Repository Reference Disposal Concepts and Thermal Load Management Analysis*. FCRD-UFD-2012-00219 Rev. 2. U.S. Department of Energy, Office of Used Fuel Disposition.

Havlova, V., A. Laciok, D. Volpalka, and M. Andrlik. 2006. "Geochemical Study of Uranium Mobility in Tertiary Argillaceous System at Ruprechtov Site, Czech Republic, Czechoslovak." *Journal of Physics* 56: Supplement D, D81-D86.

Hedin, A., E. Andersson, J. Andersson, C. Greis, L.Z. Evins, U. Kautsky, C. Lilja, T. Lindborg, M. Lindgren, M. Lofgren, R. Munier, J.-O. Naslund, I. Puigdomenech, P. Sellin, J.-O. Selroos, K. Skagius, K. Spahiu and F. Vahlund. 2011. The SR-Site Safety Assessment for Licensing a Spent Fuel Repository in Sweden. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011,* Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

International Atomic Energy Agency. 1997. *Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.* INFCIRC/546.

International Atomic Energy Agency. 1999a. *Inventory of Radioactive Waste Disposals at Sea.* IAEA-TECDOC-1105.

International Atomic Energy Agency. 1999b. *Use of Natural Analogues to Support Radionuclide Transport Models for Deep Geological Repositories for Long Lived Radioactive Wastes.* TECDOC-1109.

International Atomic Energy Agency. 2003a. *Scientific and Technical Basis for Geological Disposal of Radioactive Wastes*. Technical Reports Series No. 413.

International Atomic Energy Agency. 2003b. *Reference Biospheres for Solid Radioactive Waste Disposal.* IAEA-BIOMASS-6. Vienna, Austria: International Atomic Energy Agency.

International Atomic Energy Agency. 2006. Geological Disposal of Radioactive Waste. IAEA Safety Standards Series No. WS-R-4. Vienna, Austria: International Atomic Energy Agency and the Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

International Atomic Energy Agency. 2011. *Disposal of Radioactive Waste, Specific Safety Requirements.* IAEA Safety Standards Series No. SSR-5. Vienna, Austria: International Atomic Energy Agency.

Jantzen, C.M., D.I. Kaplan, N.E. Bibler, D.K. Peeler and M.J. Plodinec. 2008. "Performance of a buried radioactive high level waste (HLW) glass after 24 years." *Journal of Nuclear Materials* (378): 244–256.

Kremer, E.P., J.D. Avis, T. Chshyolkova, F, Garisto, P. Gierszewski, M. Gobien, N.G. Hunt, C.I. Kitson, C.L.D. Medri, T.W. Melnyk, L.C. Wojciechowski. 2011. Postclosure Safety Assessment of a Deep Geological Repository for Canada's Used Nuclear Fuel. ISBN: 978-0-89448-085-0. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011,* Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

Lappin, A.R., R.L. Hunter, D.P. Garber, and P.B. Davies (eds.). 1989. *Systems Analysis, Long-Term Radionuclide Transport, and Dose Assessments, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico; March 1989*. SAND89-0462. Albuquerque, NM: Sandia National Laboratories.

Law, J., D. Peterman, C. Riddle, D. Meikrantz and T. Todd. 2007. Advances in Development of the Fission Product Extraction Process for the Separation of Cesium and Strontium from Spent Nuclear Fuel. OSTI ID 918190. *11th International Conference on Environmental Remediation and Radioactive Waste Management (ICEM 2007)*, Bruges, Belgium (September 2-6, 2007).

Lee, J.O., W.J. Cho, P.S. Hahn, and H.H. Park. 1994. Diffusion Characteristics of Iodide in a Domestic Bentonite of Korea. *Journal of the Korean Nuclear Society* 26: 285-293.

Levich, R.A and J. Stuckless. 2007. *Yucca Mountain, Nevada – A Proposed Geologic Repository for High-Level Radioactive Waste,* in: eds: J.S. Stuckless and R.A. Levich, *The Geology and Climatology of Yucca Mountain and Vicinity, Southern Nevada and California.* Memoir 199. Geological Society of America.

Lutze, W. 2006. “Glass” in: (R. Ewing and A. Macfarlane, eds.) *Uncertainty Underground: Yucca Mountain and the Nation’s High-Level Nuclear Waste*. Cambridge, MA: MIT Press.

Maes, A., C. Bruggeman, and J. Vancluysen. 2004. Reduction and Solubility Behaviour of Selenium under Boom Clay Conditions. EUR-21027. *Proceedings: Euradwaste'04. Radioactive Waste Management Community Policy and Research Initiatives.*

Mariner, P.E., J.H. Lee, E.L. Hardin, F.D. Hansen, G.A. Freeze, A.S. Lord, B. Goldstein, and R.H. Price. 2011. *Granite Disposal of U.S. High-Level Radioactive Waste*. SAND2011-6203. Albuquerque, NM: Sandia National Laboratories.

Massachusetts Institute of Technology. 2006. *The Future of Geothermal Energy*. Cambridge, MA.

McCartin, T. 2010. *Summary Statement: Regulations for Geological Disposal of High-Level Radioactive Waste.* Presented to Blue Ribbon Commission on America’s Nuclear Future, Disposal Subcommittee, September 1, 2010.

McCartin, T. 2012. *United States Nuclear Waste Technical Review Board, Spring Board Meeting.* Transcript from March 7, 2012.

McKinley, I.G., and A. Scholtis. 1992. Compilation and comparison of radionuclide sorption databases used in recent performance assessments. Radionuclide Sorption Safety Evaluation Perspectives. *NEA Workshop.* Organisation for Economic Co-operation and Development. Paris, France, 21-55.

McKinley, I.G. and A. Scholtis. 1993. “A Comparison of Radionuclide Sorption Databases Used in Recent Performance Assessments.” *Journal of Contaminant Hydrology* 13: 347-363.

McKinley, I.G., F.B. Neall, H. Kawamura, and H. Umeki. 2006. "Geochemical optimisation of a disposal system for high-level radioactive waste." *Journal of Geochemical Exploration* 90: 1-8.

McMurry, J., D.A. Dixon, J.D. Garroni, B.M. Ikeda, S. Stroes-Gascoyne, P. Baumgartner, and T.W. Melnyk. 2003. *Evolution of a Canadian deep geologic repository: Base scenario*. Nuclear Waste Management Division Report 06819-REP-01200-10092-R00. Toronto, Canada: Ontario Power Generation.

Meacham, P.G., D.R. Anderson, E.J. Bonano, and M.G. Marietta. 2011. *Sandia National Laboratories Performance Assessment Methodology for Long-Term Environmental Programs: The History of Nuclear Waste Management*. SAND2011-8270. Albuquerque, NM: Sandia National Laboratories.

Miller, W., R. Alexander, N. Chapman, I. McKinley, and J. Smellie. 1994. “Natural Analogue Studies in the Geological Disposal of Radioactive Wastes.” *Studies in Environmental Science* **57**. Amsterdam: Elsevier.

Miller, W., R. Alexander, N. Chapman, I. McKinley, and J. Smellie. 2000. *Geological Disposal of Radioactive Wastes & Natural Analogues.* Waste Management Series, Volume 2. Amsterdam: Pergamon.

Möller, P., S.M. Weise, E. Althaus, W. Bach, H.J. Behr, R. Borchardt, K. Bräuer, J. Drescher, J. Erzinger, E. Faber, B.T. Hansen, E.E. Horn, E. Huenges, H. Kämpf, W. Kessels, T. Kirsten, D. Landwehr, M. Lodemann, L. Machon, A. Pekdeger, H.-U. Pielow, C. Reutel, K. Simon, J. Walther, F.H. Weinlich, and M. Zimmer. 1997. Paleofluids and Recent Fluids in the Upper Continental Crust: Results from the German Continental Deep Drilling Program (KTB). J. Geophys. Res. 102(B8): 18233–18254.

Montazer, P. and W.E. Wilson. 1984. *Conceptual Hydrologic Model of Flow in the Unsaturated Zone, Yucca Mountain, Nevada*. Water-Resources Investigations Report 84-4345. Lakewood, CO: U.S. Geological Survey.

Muller, A.B., N.C. Finley, and J. Pearson, Jr. 1981. *Geochemical Parameters used in the Bedded Salt Reference Repository Risk Assessment Methodology*. NUREG/CR-1996; SAND0557. Albuquerque, NM: Sandia National Laboratories.

Nagra (National Cooperative for the Disposal of Radioactive Waste). 2002. *Project Opalinus Clay: Safety Report, Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis)*. Technical Report 02-05. Wettingen, Switzerland: Nagra.

National Academy of Sciences. 1957. *The Disposal of Radioactive Waste on Land*; *Report of the Committee on Waste Disposal of the Division of Earth Sciences.* Publication 519. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Academy of Sciences. 1978. *Geological Criteria for Repositories for High-Level Radioactive Wastes*. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Academy of Sciences. 1983. *A Study of the Isolation Systems for Geologic Disposal of Radioactive Wastes*. Waste Isolation Systems Panel, Board on Radioactive Waste Management, Commission on Physical Sciences, Mathematics, and Resources. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Academy of Sciences. 1995. *Technical Bases for Yucca Mountain Standards.* Committee on Technical Bases for Yucca Mountain Standards, Board on Radioactive Waste Management. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Academy of Sciences. 1996. *The Waste Isolation Pilot Plant: A Potential Solution for the Disposal of Transuranic Waste*. ISBN: 0-309-58923-1. Committee on the Waste Isolation Pilot Plant, National Research Council. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press. http://www.nap.edu/catalog/5269.html.

National Academy of Sciences. 2001. *Disposition of High-Level Waste and Spent Nuclear Fuel: The Continuing Societal and Technical Challenges*. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Academy of Sciences. 2003. *One Step at a Time: The Staged Development of Geologic Repositories for High-Level Radioactive Waste*. National Research Council, Board on Radioactive Waste Management. Washington, DC: National Academy of Sciences–National Research Council, National Academy Press.

National Atlas. 2012. *Map Maker.* www.nationalatlas.gov, accessed January 3, 2012.

Neuzil, C.E. 2000. “Osmotic generation of ‘anomalous’ fluid pressures in geologic environments.” *Nature* 403: 182-184.

Nuclear Waste Policy Act. 1983a. *Public Law 97-425; 96 Stat. 2201, as amended by Public Law 100-203, Title I. December 22, 1987*.

Office of Crystalline Repository Development. 1983. *A National Survey of Crystalline Rocks and Recommendations of Regions to be Explored for High-Level Radioactive Waste Repository Sites.* Technical Report OCRD-1.Battelle Memorial Institute.

Office of Technology Assessment. 1985. *Managing the Nation's Commercial High-Level Radioactive Waste.* Congress of the United States, New York: UNIPUB InfoSource International.

ONDRAF/NIRAS (Belgian Agency for Radioactive Waste and Enriched Fissile Materials). 2001. *Technical Overview of SAFIR 2: Safety Assessment and Feasibility Interim Report 2.* NIROND 2001-05 E. Belgium: ONDRAF/NIRAS.

Organisation for Economic Co-operation and Development. 1992. *Systematic Approaches to Scenario Development: A Report of the NEA Working Group on Identification and Selection of Scenarios for Performance Assessment of Radioactive Waste Disposal*. Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-operation and Development. 1995. *The Environmental and Ethical Basis of Geological Disposal of Long-Lived Radioactive Wastes*. Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-operation and Development. 1996. *Water, Gas and Solute Movement through Argillaceous Media*. Nuclear Energy Agency Report CC-96/1. Prepared by S. T. Horseman, J.J.W. Higgo, J. Alexander, and J. F. Harrington. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-operation and Development. 1999a. *An International Database of Features, Events and Processes*. Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-operation and Development. 1999b. *Confidence in the Long-Term Safety of Deep Geological Repositories: Its Development and Communication*. Vienna, Austria: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-Operation and Development. 2003a. *Engineered Barrier Systems and the Safety of Deep Geological Repositories.* State of the Art Report, EUR 19964 EN. Paris, France: Organisation for Economic Co-operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-Operation and Development. 2003b. *Features, Events and Processes Evaluation Catalogue for Argillaceous Media*. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-Operation and Development. 2004. *Post-Closure Safety Case for Geological Repositories Nature and Purpose.* Nuclear Energy Agency No. 3679, ISBN 92-64-02075-6. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-Operation and Development. 2006. *The NEA International FEP Database: Version 2.1*. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency. http://www.nea.fr/rwm/documents/NEAFEP2006.zip

Organisation for Economic Co-Operation and Development. 2007. Safety Cases for Deep Geological Disposal of Radioactive Waste: Where Do We Stand? *Symposium Proceedings*, Paris, France (January 23-25, 2007).

Organisation for Economic Co-operation and Development. 2008. *Moving Forward with Geological Disposal of High-Activity Radioactive Waste, A Collective Statement by the NEA Radioactive Waste Management Committee (RWMC).* NEA No. 6433. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency.

Organisation for Economic Co-operation and Development. 2012. *Methods for Safety Assessment of Geological Disposal Facilities for Radioactive Waste, Outcomes of the NEA MeSA Initiative.* NEA No.

6923. Paris, France: Organisation for Economic Co-Operation and Development, Nuclear Energy Agency.

Palmer, R.A., H.M. Houston and A.J. Misercola. 2004. *Completion of the Vitrification Campaign at the West Valley Demonstration Project, Environmental Issues and Waste Management Technologies IX*. J.D. Vienna and D.R. Spearing (Eds.), *Ceramic Transactions* 155: 179-196.

Park, Y.-J., E.A. Sudicky, and J.F. Sykes. 2009. Effects of Shield Brine on the Safe Disposal of Waste in Deep Geologic Environments. *Advances in Water Resources* 32: 1352-1358.

Patrick, W.C. 1986. *Spent-Fuel Test—Climax: An Evaluation of the Technical Feasibility of Geologic Storage of Spent Nuclear Fuel in Granite, Executive Summary of Final Results*. UCRL-53762. University of California, Lawrence Livermore National Laboratory.

Pecos Management Services. 2010 *Assessment of an Alternative Approach for Remote Handled Transuranic Waste Disposal*. ISO-2 Project WIPP Independent Oversight – DE-AC30-06EW03005.

Posiva. 2006. *Expected Evolution of a Spent Nuclear Fuel Repository at Olkiluoto.* B. Pastina and P. Hellä (eds.). POSIVA 2006-05. Eurajoki, Finland: Posiva Oy.

Posiva. 2008. *Groundwater Flow and Transport Simulations in Support of RNT-2008 Analysis.* Working Report 2008-52. Eurajoki, Finland: Posiva Oy.

Posiva. 2010a. *Interim Summary Report of the Safety Case 2009.* POSIVA 2010-02, ISBN 978-951-652-173-5. Eurajoki, Finland: Posiva Oy.

Posiva. 2010b. *Models and Data Report 2010*. POSIVA 2010-01, ISBN 978-951-652-172-8. Eurajoki, Finland: Posiva Oy.

Rechard, R.P., B. Goldstein, L.H. Brush, J.A. Blink, M. Sutton, and F.V. Perry. 2011. *Basis for Identification of Disposal Options for Research and Development for Spent Nuclear Fuel and High-Level Waste.* FCRD-USED-2011-000071, SAND2011-3781P. Albuquerque, NM: Sandia National Laboratories.

Ruiz Lopez, C., J. Rodriguez, P. Hernan, F. Recreo, C. Ruiz, P. Prado, M.J. Gimeno, L.F. Auque, J. Gomez, P. Acero, A. Gonzalez, J. Samper, L. Montenegro, J. Molinero, J. Delgado, A. Criado, J.A. Martinez, and S. Ruiz. 2004. *Analogue Application to Safety Assessment and Communication of Radioactive Waste Disposal: Illustrative Synthesis*. Colecccion: Documentos I+D, Referencia: DID-11.04. ENRESA.

Sandia National Laboratories. 2007a. Drift-Scale THC Seepage Model. MDL-NBS-HS-000001, REV 05. Las Vegas, NV: Sandia National Laboratories.

Sandia National Laboratories. 2007b. *Waste Form and In-Drift Colloids-Associated Radionuclide Concentrations: Abstraction and Summary.* MDL-EBS-PA-000004 REV 03. Albuquerque, NM: Sandia National Laboratories.

Sandia National Laboratories. 2008a. *Features, Events, and Processes for the Total System Performance Assessment: Methods.* ANL-WIS-MD-000026 REV 00. Las Vegas, NV: Sandia National Laboratories.

Sandia National Laboratories. 2008b. *Features, Events, and Processes for the Total System Performance Assessment: Analyses.* ANL-WIS-MD-000027 REV 00 ACN 01. Las Vegas, NV: Sandia National Laboratories.

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

Sandia National Laboratories. 2010. *US/German Workshop on Salt Repository Research, Design, and Operation, May 25–27, 2010.* http://www.sandia.gov/SALT/SALT_Home.html.

Schulz, W.W. and E.P. Horwitz. 1988. "The TRUEX Process and the Management of Liquid TRU Waste." *Separation Science and Technology* **23**(12 & 13): 1191-1210.

Schwyn, B. and P. Wersin. 2004. The Compilation of Geochemical Data Bases for Performance Assessment of a Planned Repository in the Opalinus Clay in Northern Switzerland. *Material Research Society Symposium Proceedings* 807: 1-6.

Sevougian, S.D., G.A. Freeze, M.B. Gross, J. Lee, C.D. Leigh, P. Mariner, R.J. MacKinnon, and P. Vaughn. 2012. *GTSPA Model Development and Sensitivity Analysis of Processes Affecting Performance of a Salt Repository for Disposal of Heat-Generating Nuclear Waste*. FCRD-USED-2012-000320, Rev. 0. U.S. Department of Energy, Office of Used Fuel Disposition.

Siegmann, E.R. 2006. "Zircaloy Cladding" in: (R. Ewing and A. Macfarlane, eds.) *Uncertainty Underground: Yucca Mountain and the Nation's High-Level Nuclear Waste*. Cambridge, MA: MIT Press.

SKB (Swedish Nuclear Fuel and Waste Management Company). 2006a. *Data Report for the Safety Assessment SR-Can*. Technical Report TR-06-25. Stockholm, Sweden: SKB.

SKB. 2006b. *Long-term safety for KBS-3 repositories at Forsmark and Laxemar – a first evaluation: Main report of the SR-Can project.* Technical Report TR-06-09. Stockholm, Sweden: SKB. www.skb.se

SKB. 2010. *Buffer, Backfill and Closure Process Report for the Safety Assessment SR-Site.* Technical Report TR-10-47. Stockholm, Sweden: SKB.

SKB. 2011. *Long-term Safety for the Final Repository for Spent Nuclear Fuel at Forsmark: Main Report of the SR-Site Project.* Technical Report TR-11-01 (3 volumes). Stockholm, Sweden: SKB. www.skb.se

SKBF/KBS (Swedish Nuclear Fuel Supply Company/Division KBS). 1983. *Final Storage of Spent Nuclear Fuel – KBS-3.* Volume 1, General. Stockholm, Sweden: SKBF/KBS.

Smedes, H.W. 1980. *Rationale for the Geologic Isolation of High-Level Radioactive Waste, and the Assessment of the Suitability of Crystalline Rocks.* Open-File Report 80-1065. U.S. Geological Survey.

Soelberg, N.R., S.J. Piet, S.E. Bays, R.N. Henry, D.H. Meikrantz, T.A. Taiwo, T.K. Kim, D.Yun, J. Gehin, L. Qualls, K. Williams and W. Halsey. 2010. *Waste Stream Analyses for Nuclear Fuel Cycles*. INL/EXT-10-19643. Idaho Falls, ID: Idaho National Laboratory, Fuel Cycle Research & Development. August 2010.

Strachan, D.M., S.A. Bryan, C.H. Henager, T.G. Levitskaia, J. Matyas, P.K. Thallapally, R.D. Scheele, W.J. Weber and F. Zheng. 2009. *Processes for Removal and Immobilization of* ^{14}C, ^{129}I, *and* ^{85}Kr. PNNL-18852. Richland, WA: Pacific Northwest National Laboratory.

Swift, P. 2011. *Basis for Identification of Disposal Options for Research and Development.* Presentation to U.S. Nuclear Waste Technical Review Board.

Swift, P.N., B.W. Arnold, P.V. Brady, G. Freeze, T. Hadgu, J.H. Lee, and Y. Wang. 2011. Preliminary Performance Assessment for Deep Borehole Disposal of High-Level Radioactive Waste. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011,* Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

Thegerström, C. and O. Olsson. 2011. The License Application for the KBS-3-System - A Milestone for the Swedish Spent Fuel Disposal Program. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011,* Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

Tien, P.L., F. Nimick, A. Muller, P. Davis, R. Guzowski, L. Duda, and R. Hunter. 1983. *Repository Site Data and Information in Bedded Salt: Palo Duro Basin, Texas*. NUREG/CR-3129, SAND82-2223. Albuquerque, NM: Sandia National Laboratories.

Tribal Energy and Environmental Information. 2012. *Oil and Gas Resources in the United States.* Office of Indian Energy and Economic Development. http://teeic.anl.gov/er/oilgas/restech/dist/index.cfm, accessed January 3, 2012. NOTE: Data for map from Mast, R.F., D.H. Root, L.P. Williams, and W.R. Beeman. 1995. *Areas of historical oil and gas exploration and production in the conterminous United States*. Open-File Report, 95-75-C.

U.S. Department of Energy. 1978. *Report of Task Force for Review of Nuclear Waste Management,* DOE/ER-0004/D, Washington, DC.

U.S. Department of Energy. 1979. *Report to the President by the Interagency Review Group on Nuclear Waste Management.* TD-29442.

U.S. Department of Energy. 1980. *Environmental Impact Statement on Management and Disposal of Commercially Generated Radioactive Wastes.* DOE/EIS-0046.

U.S. Department of Energy. 1981. *Program of Research and Development for Management and Disposal of Commercially Generated Radioactive Wastes; Record of Decision.* 46 FR 26677.

U.S. Department of Energy. 1986a. *Draft Area Recommendation Report for the Crystalline Repository Project: Overview.* DOE/CH-15(0). Argonne, IL: Office of Civilian Radioactive Waste Management, Crystalline Repository Project Office.

U.S. Department of Energy. 1986b. *Environmental Assessment, Davis Canyon Site.* DOE/RW-0071. Washington, DC: Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1986c. *Environmental Assessment, Deaf Smith County Site, Texas.* DOE/RW-0069. Washington, DC: Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1986d. *Environmental Assessment, Richton Dome Site, Mississippi.* DOE/RW-0072. Washington, DC: Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1986e. *Environmental Assessment, Reference Repository Location, Hanford Site.* DOE/RW-0070. Washington, DC: Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1986f. *Environmental Assessment, Yucca Mountain Site, Nevada Research and Development Area, Nevada.* DOE/RW-0073. Washington, DC: Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1986g. *A Multiattribute Utility Analysis of Sites Nominated for Characterization for the First Radioactive-Waste Repository: A Decision Aiding Methodology*. DOE/RW-0074.

U.S. Department of Energy. 1986h. *Recommendation by the Secretary of Energy of Candidate Sites for Site Characterization for the First Radioactive-Waste Repository*. DOE/RW-0048.

U.S. Department of Energy. 1987. *Site Characterization Plan Conceptual Design Report for a High-level Nuclear Waste Repository in Salt, Horizontal Emplacement Mode: Volumes 1 and 2.* DOE/CH/46656-14(1) and OSTI ID: 6960510 for Volume 1; DOE/CH/46656-15(2) and OSTI ID: 70516545 for Volume 2. Irvine, CA: Fluor Technology, Inc.

U.S. Department of Energy. 1994. *Conceptual Design for a Waste-Management System That Uses Multipurpose Canisters*. DOE/RW-0432, OSTI ID 10122484. Washington, DC: U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 1996a. *Title 40 CFR Part 191 Compliance Certification Application for the Waste Isolation Pilot Plant*. DOE/CAO-1996-2184. Twenty-one volumes. Carlsbad, NM: U.S. Department of Energy, Carlsbad Area Office.

U.S. Department of Energy. 1996b. *Plutonium Recovery from Spent Fuel Reprocessing by Nuclear Fuel Services at West Valley, New York from 1966 to 1972*.

U.S. Department of Energy. 1998. *Viability Assessment of a Repository at Yucca Mountain.* DOE/RW-0508. Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 2001. *Yucca Mountain Science and Engineering Report: Technical Information Supporting Site Recommendation Consideration.* DOE/RW-0539. Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 2002a. *Calculation Method for the Projection of Future Spent Fuel Discharges*. OSTI ID 895067. U.S. Department of Energy, Office of Civilian Radioactive Waste Management.

U.S. Department of Energy. 2002b. *Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*. DOE/EIS-0250F.

U.S. Department of Energy. 2004a. *Title 40 CFR Part 191 Compliance Recertification Application for the Waste Isolation Pilot Plant.* DOE/WIPP 2004-3231. Carlsbad, NM: U.S. Department of Energy Waste Isolation Pilot Plant, Carlsbad Field Office.

U.S. Department of Energy. 2004b. *WIPP Site Selection and Characterization Studies, General Training on Methodologies for Geological Disposal in North America*. http://esd.lbl.gov/IAEAnetworktraining/session_summaries/pdfs/Sess_14_Nelson_WIPP.pdf accessed January 1, 2012. Carlsbad, NM: U.S. Department of Energy Carlsbad Field Office.

U.S. Department of Energy. 2007. *Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.* DOE/EIS-0250F-S1.

U.S. Department of Energy. 2008a. *The Report to the President and Congress by the Secretary of Energy on the Need for a Second Repository*. DOE/RW-0595.

U.S. Department of Energy. 2008b. *Yucca Mountain Repository License Application Safety Analysis Report*. DOE/RW-0573, Revision 1. http://www.nrc.gov/waste/hlw-disposal/yucca-lic-app/yucca-lic-app-safety-report.html#1.

U.S. Department of Energy. 2009. *Title 40 CFR Part 191 Subparts B and C Compliance Recertification Application for the Waste Isolation Pilot Plant.* DOE/WIPP-09-3424. Carlsbad, NM: DOE.

U.S. Department of Energy. 2011a. *10,000th Shipment Celebrated at WIPP*. http://www.wipp.energy.gov/pr/2011/10,000th_Shipment_Commemoration_Release_Final.pdf, accessed Dec 30, 2011.

U.S. Department of Energy. 2011b. *A Management Proposal for Salt Disposal Investigations with a Field Scale Heater Test at WIPP*. DOE/CBFO-11-3470, Rev. 0. Carlsbad, NM: U.S. Department of Energy, Carlsbad Field Office.

U.S. Department of Energy. 2012. *Used Fuel Disposition Campaign Disposal Research and Development Roadmap.* FCR&D-USED-2011-000065 REV 1.

U.S. Department of Transportation. 2009. *Technical Manual for Design and Construction of Road Tunnels – Civil Elements.* Federal Highway Administration Publication No. FHWA-NHI-10-034. http://www.fhwa.dot.gov/bridge/tunnel/pubs/nhi09010/index.cfm.

U.S. Environmental Protection Agency. 1985. *Background Information Document, Final Rule for High-Level and Transuranic Radioactive Wastes*. *Environmental Standards for the Management and Disposal* *of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste*. 40 CFR Part 91, EPA 520/1-85-023. U.S. Environmental Protection Agency.

U.S. Geological Survey 1980. *Plan for Identification and Geological Characterization of Sites for Mined Radioactive Waste Repositories.* Open-File Report 80-686. Water-Resources Investigations.

U.S. Nuclear Regulatory Commission. 2003. *Yucca Mountain Review Plan.* Final Report. NUREG-1804, Rev. 2. Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards.

U.S. Nuclear Regulatory Commission. 2011a. *About NRC: History.* http://www.nrc.gov/about-nrc/emerg-preparedness/history.html, accessed December 21, 2011.

U.S. Nuclear Regulatory Commission. 2011b. *Map of Power Reactor Sites.* http://www.nrc.gov/reactors/operating/map-power-reactors.html, accessed January 18, 2012.

U.S. Nuclear Waste Technical Review Board. 2009. *Survey of National Programs for Managing High-Level Radioactive Waste and Spent Nuclear Fuel: A Report to Congress and the Secretary of Energy*. http://www.nwtrb.gov/reports/nwtrb_survey_rpt.pdf

U.S. Nuclear Waste Technical Review Board. 2011. *Technical Advancements and Issues Associated with the Permanent Disposal of High Activity Wastes, Lessons Learned from Yucca Mountain and Other Programs*. June.

Vietti-Cook, A.L. 1997. *Staff Requirements - Comsecy-96-061, Risk-Informed, Performance-Based Regulation (DSI 12).* Memorandum to L. Joseph Callan, Executive Director for Operations, U.S. Nuclear Regulatory Commission. April 15.

Volckaert, G., L. Ortiz, P. de Canière, M. Put, S.T. Horseman, J.F. Harrington, V. Fioravante, and M. Impey 1995. MEGAS: Modelling and Experiments on Gas Migration in Repository Host Rock. 447 pp. EUR-16235, Luxemburg, E.C.

Walker, J.S. 2009. *The Road to Yucca Mountain: The Development of Radioactive Waste Policy in the United States.* University of California Press, Berkeley, CA, Chapter 3, *An Atomic Garbage Dump for Kansas.*

Wang, Y., and J. Lee (eds.). 2010. *Generic Disposal System Environment Modeling--Fiscal Year 2010 Progress Report*. Prepared for U.S. Department of Energy, Office of Nuclear Energy, Fuel Cycle Research and Development Program. Albuquerque, NM: Sandia National Laboratories.

Wantz, O., A. Dubus, and R. Beauwens. 2004. Belgian Supercontainer Long-Term Criticality Calculations. Paper WM-4202l. Waste Management (WM) 2004 Conference, Phoenix, AZ (December 4-6, 2004).

Weaver, B. and F.A. Kappelmann. 1964. *TALSPEAK: A New Method of Separating Americium and Curium from the Lanthanides by Extraction from an Aqueous Solution of an Aminopolyacetic Acid Complex with a Monoacidic Organophosphate or Phosphonate*. ORNL-3559. Oak Ridge, TN: Oak Ridge National Laboratory.

Weber, J.R., S. Keller, S. Mrugalla, J. Wolf, D. Buhmann, J. Mönig, W. Bollingerfehr, J. Krone, A. Lomerzheim. 2011. Safety Strategy and Assessment for a German HLW-Repository in Salt. *Proceedings of the 13th International High-Level Radioactive Waste Management Conference (IHLRWMC) 2011*, Albuquerque, NM (April 10-14, 2011). American Nuclear Society.

Weiss, W. (Chair), C.-M. Larsson, Chr. McKenney, J.-P. Minon, S. Mobbs, T. Schneider, H. Umeki, Corresponding members W. Hilden, C. Pescatore, and M. Vesterind. 2011. *Radiological Protection in Geological Disposal of Long-Lived Solid Radioactive Waste.* Draft Report for Consultation. ICRP ref 4838-8963-9177. Annals of the ICRP.

Appendix A

History of Repository Siting in the U.S.

Appendix A—History of Repository Siting in the U.S.

A-1. Introduction

The history of repository siting in the United States dates to the mid-1950's when the country began to develop commercial nuclear power and sought a solution for the disposal of the wastes arising from the reprocessing of nuclear fuels. At the request of the Atomic Energy Commission a committee of geologists and geophysicists was established by the National Research Council of the National Academy of Sciences to consider the possibilities of disposing of high-level radioactive wastes in quantity within the continental limits of the United States. The Committee was convinced that radioactive waste could be disposed of safely in a variety of ways and at a large number of sites in the United States (National Academy of Sciences 1957). This Appendix presents an overview of the history of repository siting in the U.S. The start of nuclear waste management activities in the U.S. began under the Atomic Energy Commission, the predecessor agency to both the Department of Energy and the Nuclear Regulatory Commission. About the time the Energy Research and Development Administration was created from the Atomic Energy Commission in 1974, the search for sites for development of a repository in the U.S. began to develop a focus with the early efforts of the National Waste Terminal Storage Program. It was not until passage of the Nuclear Waste Policy Act in 1982, however, that the search for sites for development of a repository adopted a formal structure defined by law. The Nuclear Waste Policy Act Program proceeded until 1987, when Congress amended the Nuclear Waste Policy Act and selected a single site to be studied. In response to direction provided in the amended Nuclear Waste Policy Act, the Secretary of Energy produced the Report to Congress on the Need for a Second Repository (U.S. Department of Energy 2008a); that report contained a summary of sites and areas that had been looked at previously in the U.S. in the various historical efforts to identify potential sites for repository development.

There are a number of excellent references available that document in detail the history of repository siting in the U.S. (see, for example, Carter 1987).

A-2. The Start of Nuclear Waste Management Activities in the U.S.

President Eisenhower gave his Atoms for Peace speech to the United Nations in 1953; commercial nuclear power generation was a cornerstone of his plan. The Atomic Energy Commission was assigned responsibility by the Atomic Energy Act (Atomic Energy Act 1954) for managing used nuclear fuel from civilian reactors. The Shippingport Atomic Power Station, which began operating in 1957, would become the first operational commercial nuclear power plant in the U.S. It was also the first in the world devoted solely to generating commercial power. (U.S. Nuclear Regulatory Commission 2011a).

The search for permanent storage for radioactive waste began in 1955 when the Atomic Energy Commission asked the National Academy of Sciences to examine the issue of what to do with the wastes from the fuel cycle, under the assumption that the wastes would be dissolved in very low concentration in liquid (Office of Technology Assessment 1985). At that time, the country was reprocessing used nuclear fuel in order to be able to reuse the uranium remaining in the used nuclear fuel and obtain plutonium for weapons. In 1957 the National Academy of Sciences reported ". . . that radioactive waste can be disposed of safely in a variety of ways and at a large number of sites in the United States." The Academy also indicated that ". . . the most promising method of disposal of high-level waste . . . is in salt deposits" (National Academy of Sciences 1957). Deep geologic disposal in salt formations was seen as a promising method to explore for disposing of high-activity waste without aging the waste to lower the thermal load. With the technology currently available in the 1950 and 1960s, the Atomic Energy Commission gave mined disposal in salt priority. However, the Atomic Energy Commission was slow in implementing a solution (Rechard et al. 2011).

Other situations in the U.S. gave rise to pressures to develop a repository. In May 1969, the Rocky Flats Plant, a weapons component production facility caught fire. Located only 26 km from Denver, the fire and subsequent cleanup attracted public attention. The press reported that radioactive waste from the cleanup was to be sent to Idaho. The public and many state officials learned that waste from Rocky Flats had routinely been sent to the Radioactive Waste Management Complex in Idaho; located at the Idaho National Laboratory reservation near the Snake River and its associated aquifer, it was viewed as a less than ideal location for permanent disposal of the nuclear wastes. The Atomic Energy Commission sought a more suitable site, and in June 1970, tentatively selected the abandoned Carey salt mine near Lyons, Kansas, the site of an underground research laboratory in salt studying heat dissipation operated by Oak Ridge National Laboratory between 1963 and 1967 (Walker 2009).

Although salt has many advantages, a disadvantage is the frequent coexistence of economic minerals and hydrocarbons. In 1971, a large number of boreholes for mineral exploration and some solution mining were discovered near the proposed mine near Lyons, Kansas. Because of local opposition, the Atomic Energy Commission abandoned the Lyons Project and announced to Congress in May 1972 plans for a Retrievable Surface Storage Facility, in which waste could be stored “a minimum of 100 years” and enable the U.S. to “keep open all options” and to “move slowly” to permanent disposition (Rechard et al. 2011).

The Environmental Protection Agency and others, through comments on the Environmental Impact Statement for a Retrievable Surface Storage Facility issued in 1974, claimed such a facility (possibly located at the Hanford reservation, the Idaho Radioactive Waste Management Complex, or the Nevada Test Site) was de facto permanent disposal. In contrast, some commenters thought a Retrievable Surface Storage Facility would take pressure off finding a disposal site (Office of Technology Assessment 1985). The criticism prompted the newly formed Energy Research and Development Administration, the agency charged with implementing nuclear power following the dismantling of the Atomic Energy Commission, to abandon surface storage as a near term solution and emphasize the search for disposal sites.

In the late 1970s, the United States established the policy of not reprocessing commercial used nuclear fuel over concerns for proliferation risks (see, for example, Carter1987). This had the effect of changing the focus of the program to direct disposal of used nuclear fuel.

In 1972, the Atomic Energy Commission asked the U.S. Geological Survey to evaluate several different methods of geologic disposal, principally in geologic media other than salt. Five modes of disposal were to be considered:

1. Very deep drill holes
2. Geometric array of shallow to moderate depth drill holes
3. Shallow mined chambers
4. Cavities with manmade (engineered) barriers
5. Explosion cavities

The final report (Ekren et al. 1974) cited 30 previous reports on geologic disposal and concluded that hydrologic isolation was of paramount importance. One specific recommendation was "the Basin and Range Province of the western United States, particularly the Great Basin exclusive of seismic-risk zone 3,” appears to have potential for mined chambers above deep water tables in tuff, shale, or argillite.” This opened the option of locating waste above the water table, in the unsaturated zone, rather than beneath the water table.

The Atomic Energy Commission published its first technical analysis of methods for long-term management of used nuclear fuel and high-level radioactive waste in 1974 along with a summary of high-activity waste management alternatives; an expanded update, was published in 1976. The reports described options, but did not present conclusions or policy recommendations. The generation of options was thorough in that no new categories of options have been identified since, although a few concepts were later added to some of the categories. For example, disposal in volcanoes or magma chambers was proposed in the early 1970s and the idea was again proposed recently; however, a reliable method to get close enough to emplace the waste has not been proposed nor has the fate after emplacement been modeled (Rechard et al. 2011).

Throughout the 1970s and 1980s, in addition to working with the U.S. Department of Energy in specific areas, the U.S. Geological Survey was tasked by Congress to study and comment on the problem of disposal of high-activity waste. The numerous evaluations for radioactive waste isolation had not identified options more feasible than geologic disposal and mined repositories in particular, and, between 1976 and 1980, the technical community and the U.S. Department of Energy continued to favor mined, geologic disposal. Scientists at the U.S. Geological Survey expressed confidence in geologic disposal for high-activity waste, but noted that more knowledge about the geologic barrier was needed. They also supported the concept of multiple barriers, which expanded the range of feasible geologic media for storage and disposal. The U. S. Geological Survey report (Bredehoeft et al. 1978) concluded that:

- Salt was less than ideal as a disposal medium
- Shales, tuffs, and crystalline rocks should be considered
- Major studies of flow and transport were needed, especially in fractured rock
- More tools were needed for dating water and materials older than about 40,000 years
- The severe limitations of Earth science predictions needed to be recognized

A-3. The Search for Sites for Development of a Repository in the U.S.

A-3.1 National Waste Terminal Storage Program

In 1975, the Energy Research and Development Administration began a search for possible repository sites. The National Waste Terminal Storage Program initiated efforts to screen sites in 36 States and to develop the technology for licensing, construction, operation, and closure of a repository. The sites eventually were considered for the first and second repository programs under the Nuclear Waste Policy Act, included these sites as well as sites that were previously contaminated from weapons related activities. This brought the Nevada Test Site and eventually Yucca Mountain, and the Hanford Reservation into the national screening effort (Carter 1987).

At this point the Energy Research and Development Administration had to refocus its efforts to locate a repository site. In January 1976 the Energy Research and Development Administration created the Office of Waste Isolation, which became responsible for managing the research and development aspect of the National Waste Terminal Storage Program. The Office of Waste Isolation conceived an extensive program to consider three major geologic media, salt, argillite, and crystalline rock. The initial interest in salt formations focused around one near Alpena, MI but Governor Milliken strongly objected and efforts were discontinued (Carter 1987).

The National Waste Terminal Storage Program site screening process was based on a twofold approach. The first approach focused on a survey of areas underlain by salt; the second focus on federal lands where radioactive materials were already present. Site screening was initiated at the Hanford Site and the Nevada Test Site.

In November of 1976, Energy Research and Development Administration Administrator, Robert Seamans formally outlined the detail of the National Waste Terminal Storage Program and sent letters to the Governors, Senators and Congressman of 36 states that had formations of salt, argillite, and crystalline rock that were considered potential locations for a repository. The program considered the creation of six repositories by 2000, the first two in salt, and the remaining ones in argillite and crystalline rock depending on the status of site evaluation in these newly considered geologic media. This occurred just after the election of Carter as president, but before he took office in January of 1977 (Carter 1987).

The National Waste Terminal Storage program investigated several alternative sites and rock types (Levich and Stuckless 2007). The program was continued by the U.S. Department of Energy when it replaced the Energy Research and Development Administration, following the Energy Reorganization Act of 1974. The two federal sites in this list—the Nevada Test Site and Hanford—were added to the National Waste Terminal Storage program when it was decided to look at federally controlled land that was previously contaminated by weapons related activities. The sites considered included

- **Salt Sites** (other than Lyons, Kansas)—Three salt domes (two in Mississippi and one in Louisiana) and four bedded salt units (Paradox Basin in Utah and Permian Basin of West Texas) were evaluated
- **Crystalline Rocks**—Following a survey of crystalline rocks largely in the regions of the Appalachian Mountains and the North American Shield, twelve areas in Georgia, North Carolina, Virginia, New Hampshire, Maine, Minnesota, and Wisconsin were recommended for further study
- **Sedimentary Rocks**—Widely distributed claystones and shales were considered as appropriate media for geologic disposal of nuclear waste by the National Academy of Sciences. The U.S. Department of Energy supported several investigations in this medium.
- **Basalt Waste Isolation Project, Hanford, Washington**—Investigation of layered basalts of Miocene age in the Cold Creek Syncline of the Columbia Plateau, on the Hanford Nuclear Reservation
- **Tuffaceous Rocks, Nevada Test Site**—Included tuffs in both the unsaturated and saturated zones that had been examined in considerable detail as part of other investigations on the Nevada Test Site. This site received the endorsement of United States Geological Survey Director McKelvey, who wrote to the U.S. Department of Energy in 1976 pointing out the remoteness of the site, its varied geologic environments, and the existence of 900 man-years of data collection and interpretation.

In February 1978, an internal U.S. Department of Energy task force, chaired by John Deutch, chemistry professor at the Massachusetts Institute of Technology, submitted a report on radioactive waste storage and disposal options (U.S. Department of Energy 1978) that called for more study but did note that (1) technical experts had concluded that high-level radioactive waste could be safely disposed in geologic media, (2) reprocessing was not required for safe disposal of commercial used nuclear fuel, (3) waste management should be given higher emphasis within the U.S. Department of Energy, and (4) consideration should be given to demonstrating geologic disposal of used nuclear fuel at the WIPP . Although many of the suggestions of the Deutch report were generally accepted, the suggestion to demonstrate geologic disposal of a limited number of used nuclear fuel assemblies at WIPP was very controversial in New Mexico (Carter 1987). Because of the general lack of policy guidance of the Deutch report, President Carter formed an Interagency Review Group on Nuclear Waste Management in March 1978 composed of fourteen federal agencies and also chaired by John Deutch, to propose a policy position on managing radioactive waste and the technical adequacy of geologic disposal (U.S. Department of Energy 1979).

In October 1978, the Interagency Review Group released a draft of its report for public comment that noted that "successful isolation of radioactive waste from the biosphere appears feasible for thousands of years." In March 1979, the Interagency Review Group completed its report (U.S. Department of Energy 1979) and concluded that:

- Responsibility for managing nuclear waste resides with the current generation, and in particular, the federal government
- Mined, geologic disposal was a promising method for isolating used nuclear fuel, high-level radioactive waste , and transuranic wastes such that the probability of exposure would be quite small

- Multiple barriers (the waste form and, especially, the package) were a means of compensating for geologic uncertainty
- The national program should assume that the first disposal facility would be a mined repository
- The federal government should consider a number of sites in a variety of geologies and select and build one or more intermediate scaled facilities, preferably in different regions of the U.S.
- Repository development should proceed cautiously, in a stepwise manner
- Safe interim storage should not be used as a reason to delay opening the first repository

In addition to the search for geological repositories in the United States, there was a perceived need to investigate other geologic and non-geologic means for nuclear waste disposal. The U.S. Department of Energy initiated an *Environmental Impact Statement on the Management and Disposal of Commercially Generated Nuclear Wastes* (U.S. Department of Energy 1980), which evaluated mined geologic disposal as the proposed action, and a number of alternative disposal methods, including:

- **Very Deep Hole Disposal**—Placement of waste-filled canisters in drill holes as much as 10,000 m (6 mi) deep, below circulating groundwater, and far below the accessible environment
- **Rock-melt Disposal**—Placement of waste in liquid or slurry form in a deep drill hole or underground rock opening, with the heat of radioactive decay eventually melting the surrounding rock to form a molten solution of waste and rock that would eventually solidify into a relatively insoluble mass resistant to leaching
- **Island Disposal**—Isolation of waste in a deep geologic repository beneath an uninhabited island that lies in a remote area and lacks natural resources
- **Subseabed Disposal**—Used nuclear fuel or appropriately treated high-level waste sealed in specially designed canisters and buried within deep sea sediments of an abyssal plain in a tectonically stable area far from plate boundaries
- **Ice Sheet Disposal**—Storage of waste in containers to be placed on the surface of ice sheets (such as Greenland or Antarctica), with heat from radioactive decay causing the container to melt its way toward the bottom of the ice sheet
- **Well Injection Disposal**—Injection of waste into a deep geologic formation capped by a layer of impermeable rock
- **Transmutation Concept**—Treatment methods, including reprocessing and transmutation to mitigate the waste-disposal problem also were considered
- **Space Disposal**—Several alternative concepts were considered, including (1) transport to the sun with the transport vehicle crashing into the sun’s surface, (2) emplacement of waste on the moon, and (3) sending reprocessed waste into orbit midway between Earth and Venus

The U.S. Department of Energy completed the Environmental Impact Statement in 1980. Mined geologic disposal was compared to liquid high-level radioactive waste disposal in injection wells or geologic cavities coupled with rock melt, solid disposal in deep boreholes, in the subseabed, on islands, in continental ice sheets, and in space with or without transmutation of transuranic radionuclides to faster decaying radionuclides. The Environmental Impact Statement noted, as had the Interagency Review Group, that deep borehole and subseabed disposal were worthy of further consideration. The record of decision of the Environmental Impact Statement (U.S. Department of Energy 1981) found that the least risk to mankind came from mined geologic disposal, although the differences were not significant, and were a small part of natural background radiation. Efforts to find an acceptable geologic solution were expanded. Inherent technical difficulties and other serious disadvantages limited further consideration of most of the non-geological alternatives for waste disposal.

During the deliberations of Congress on setting national policy on radioactive waste, the Congressional Office of Technology Assessment (Office of Technology Assessment 1985) concluded: The greatest single obstacle that a successful management program must overcome is the severe erosion of public confidence in the Federal Government that past problems have created. Federal credibility is questioned on three main grounds: (1) whether the Federal Government will stick to any waste policy through changes of administration, (2) whether it has the institutional capacity to carry out a technically complex and politically sensitive program over a period of decades, and (3) whether it can be trusted to respond adequately to the concerns of States and others who will be affected by the waste management program.

The United States Geological Survey published an Open File Report (Smedes 1980) on the rationale for geologic isolation of high-level radioactive waste, and assessed the suitability of crystalline rocks. That report noted that, potentially, an additional repository would be needed at a projected rate of about one every five to twenty years after the year 2000 to accommodate wastes from the projected growth of nuclear energy at that time. Further, Smedes expressed the concern that the volumes of potentially suitable salt formations and the number of suitable sites in salt were rapidly diminishing due to unanticipated resource conflicts and geologic problems in the rock. At that time a number of previously unanticipated problems of physical and chemical stability of salt as a repository medium were identified. These emerging problems, especially thermal instability, were also seen to be an issue for shale which for several years had been the second choice as a repository medium.

In contrast, granite and other crystalline rocks were seen by Smedes to have numerous favorable attributes. The report noted that there were far greater potentially suitable volumes of these rocks, for they are the most abundant rock types in the upper part of the Earth's crust. The report noted that granites were widespread in large and deep-seated homogeneous masses exposed at the surface and stable shield areas, occur in the cores of many mountain ranges, and broadly uplifted regions, and in the subsurface beneath all the younger sedimentary rock cover which in many places is thin. The report stated that compared to salt and shale, crystalline rocks have great physical strength, inherent mechanical stability, and predictable engineering characteristics that allow underground excavations to remain open for centuries. The water content of these rocks is low. At the surface and at shallow depths the water occurs in a network of fractures. In many post-tectonic crystalline rock masses these fractures largely result from decompression, as the formally deep-seated rock masses are exposed at the surface by erosion. The fractures develop at and near the surface and propagate downward in time. The report concluded that in such masses it is possible that the vertical and horizontal stresses at a depth approximating that of the planned repositories (1000 m or so) have prevented the formation of fractures, and that the rocks are virtually impermeable.

Crystalline rocks are composed of stable high temperature silicate minerals and have good sorptive properties-intermediate between shale and salt. The quantity and planning strength of any water present is usually low, thus minimizing corrosion rates and adverse effects on sorptive properties. The report noted that large volumes of crystalline rock occur in many places which have low seismic and very low tectonic activity. These attributes will help ensure that present favorable conditions will remain so during the hazardous life of the waste and in its inexorably slow movement toward the biosphere.

A-3.2 Nuclear Waste Policy Act Program

The Nuclear Waste Policy Act (1983a) endorsed the concept, identified in studies such as those of the Interagency Review Group (U.S. Department of Energy 1979), that the current generation should bear the costs of developing a permanent disposal option, and, following the U.S. Department of Energy’s Environmental Impact Statement’s Record of Decision, selected the mined geologic disposal option. The Act assigned responsibility for the waste management functions to the Office of Civilian Radioactive Waste Management, a new, single-purpose office within the U.S. Department of Energy that absorbed the functions of the National Waste Terminal Storage Program.

The Nuclear Waste Policy Act required the federal government to identify two repository sites for commercial spent nuclear fuel and high-level radioactive waste. The first repository was statutorily limited to 70,000 MTHM initially placed in reactor until a second repository was in operation. The Nuclear Waste Policy Act also defined the procedure necessary to meet its goals, and included an aggressive schedule for opening the first repository. Because of the aggressive schedule, the U.S. Department of Energy conducted site selection while developing the guidelines. In February 1983, the Secretary of Energy sent letters to the Governors of twenty-two states, notifying them that there were one or more areas within their state boundaries that were under consideration for potential sites for a repository. The alternative of starting with a new national site screening process had been explicitly considered and rejected during the debates on Nuclear Waste Policy Act (Office of Technology Assessment 1985).

Nine of the areas identified included sites that were previously under consideration by the Energy Research and Development Administration before passage of the Nuclear Waste Policy Act and some degree of preliminary site studies had been accomplished. The other potential sites had not progressed beyond area screening studies, and specific sites had not yet been identified.

The nine sites were identified for screening for the first repository under the Nuclear Waste Policy Act provisions. Shown in Figure A-1, these included three salt dome sites (one in Louisiana and two in Mississippi), four bedded salt sites (two in Texas and two in Utah), the basalt site at Hanford, Washington, and the volcanic tuff at the Nevada Test Site (U.S. Department of Energy 1986g).

NOTE: The three sites selected for characterization are shown in larger type face.

Source: Rechard et al. 2011.

Figure A-1. The Nine Sites For Which the U.S. Department of Energy Prepared Environmental Assessments and From Which Three Sites Were Nominated for Characterization under the Nuclear Waste Policy Act

By December 1984, the U.S. Department of Energy had issued draft Environmental Assessments for all nine sites; by spring of 1986, the Department had identified five of the sites as potentially acceptable for the first repository, and had issued final Environmental Assessments, in order of preference (U.S. Department of Energy 1986h) for the Yucca Mountain site in tuff in Nevada, the Richton Dome site in a salt dome in Mississippi, the Deaf Smith County site in bedded salt in Texas, the Davis Canyon site in bedded salt in Utah, and, the Hanford site in basalt in Washington.

The three sites selected for characterization were the Yucca Mountain site in tuff in Nevada, the Deaf Smith County site in bedded salt in Texas, and the Hanford site in basalt in Washington.

In addition, the U.S. had an active second-repository program considering granite formations (U.S. Department of Energy 1986a; Office of Crystalline Repository Development 1983). The U.S. Department of Energy considered bedded salt deposits, salt domes, tuff, basalt, and crystalline rock as host rock types for geologic repositories. The rock types were being analyzed at different locations within the conterminous United States. The U.S. Department of Energy considered two sources of sites for the second repository: (1) crystalline rock formations, which were already the subject of a comprehensive screening program, and (2) sites that would have been characterized for the first repository but either not selected for the first repository site or not nominated for site characterization for the first repository.

The U.S. Department of Energy evaluated geologic and environmental data for 235 crystalline rock bodies in the north-central, northeastern, and southeastern regions of the United States to identify preliminary candidate areas (Office of Crystalline Repository Development 1983); these areas are shown in Figure A-2. Further evaluation of the candidate areas resulted in the selection of twelve areas as proposed potentially acceptable sites.

Source: Office of Crystalline Repository Development 1983.

Figure A-2. Preliminary Candidate Areas Identified in the North-central, Northeastern, and Southeastern Regions of the United States

The analyses presented in the Department's Draft Area Recommendation Report (U.S. Department of Energy 1986a) demonstrated that the evidence available for each of the proposed potentially acceptable sites supported a finding that the site was not disqualified under Appendix 3 of its siting guidelines, and supported a decision to proceed with the continued investigation of the site on a basis of the favorable and potentially adverse conditions identified. The plan at that time was to finalize the Draft Area Recommendation Report and formally identify potentially acceptable sites in crystalline rock, following the Department's siting guidelines. Those potentially acceptable sites would have been investigated and evaluated in more detail during the area phase of the siting process and considered along with other candidate sites in a progressive narrowing process to finally choose the site of the second repository. The additional sites which meet the requirements for identification as potential acceptable sites under the first repository program would have retained their designation as candidate areas that the Department could identify as potentially acceptable sites if it were determined that other areas or sites were needed to meet program requirements.

Twelve proposed potentially acceptable sites, which were identified as a result of an analysis that focused on the sites having the most favorable geologic and environmental characteristics, were located in the states of Georgia, Maine (2 sites), Minnesota (3 sites), New Hampshire, North Carolina (2 sites), Virginia (2 sites), and Wisconsin. Portions of the proposed potentially acceptable sites in Wisconsin were located within the Menominee and Stockbridge-Munsee Indian reservations and portions of one of the sites in Maine were located within the Penobscot and Passamaquoddy Indian reservations. The twelve proposed potentially acceptable sites are shown in Figure A-3 (Office of Crystalline Repository Development 1983). Eight additional candidate areas are also shown.

Source: Office of Crystalline Repository Development 1983.

Figure A-3. Potentially Acceptable Sites and Candidate Areas of the Second Repository Program

A-3.3 The Secretary's Report to Congress on the Need for a Second Repository

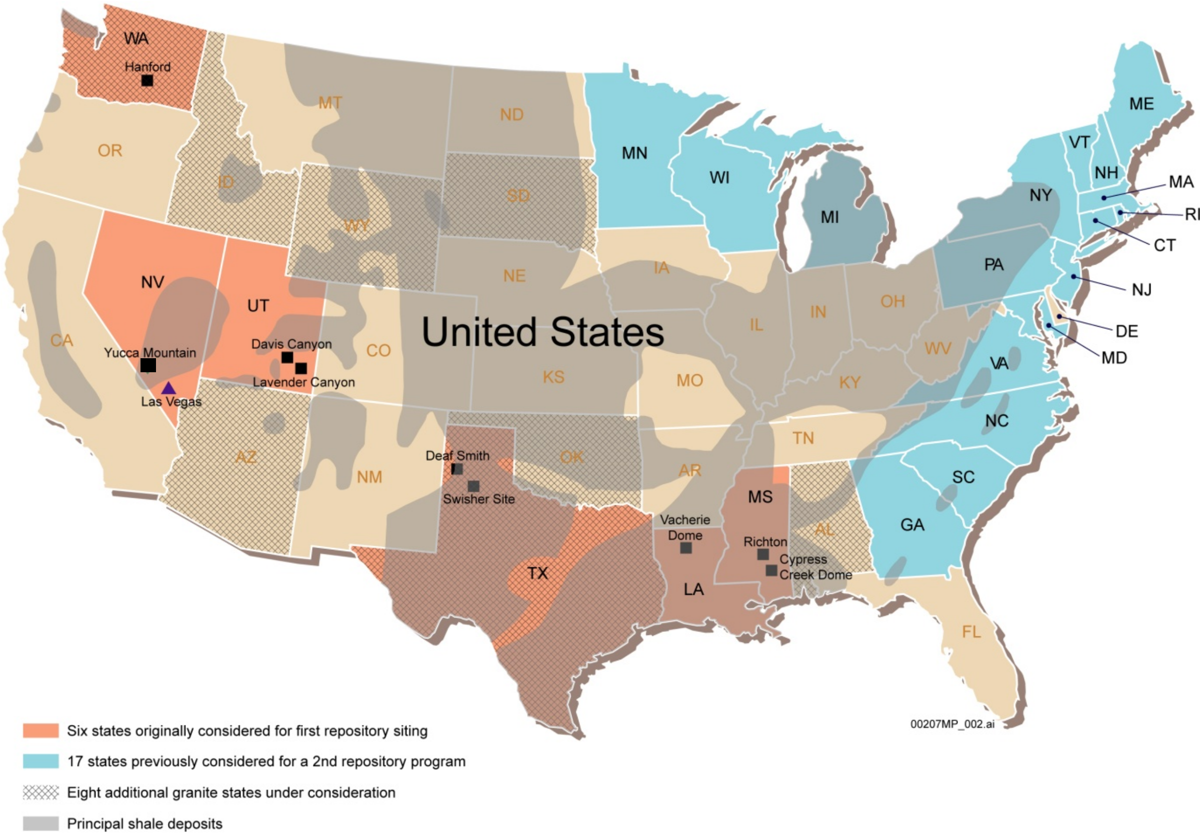
By 1986, the U.S. Department of Energy was prepared to begin to develop the site screening for the second repository program sites. There had been significant resistance to the second repository program studies and it was well expressed at public meetings regarding the program. The Secretary of Energy had concluded that there was not a pressing need to develop the second repository, and elected to postpone the program (Walker 2009). In 1987, facing rising program costs and seeking to balance the federal budget, Congress amended the Nuclear Waste Policy Act (1983a) as part of the Omnibus Budget Reconciliation Act of 1987. The amendment stopped the second repository program, and it stopped work on all of the sites in the first repository program other than Yucca Mountain. The amendment had no provision for the U.S. Department of Energy to act unilaterally if Yucca Mountain was not found to be suitable, directing the Department instead to return to Congress for further direction. The amendment also directed the Secretary report to the President and to Congress on or after January 1, 2007, but not later than January 1, 2010, on the need for a second repository (Nuclear Waste Policy Act 1983a, Sections 160 and 161(b)).

The Secretary prepared the required report (U.S. Department of Energy 2008a) and issued it to the President and Congress. The report has numerous caveats; in essence, it suggests that Congress has two choices: either lift the 70,000 MTHM statutory limit and place all of the wastes in the designated repository at Yucca Mountain, or restart the site screening processes and develop a second repository. The Secretary's Report includes a discussion of sites and areas that have been looked at during the historical screening activities in the U.S. and that could potentially be examined again.

From the first repository program, the nominations of the other two sites for possible characterization (Deaf Smith county and Hanford) remain viable options for future consideration. Draft Environmental Assessments were prepared for all nine sites identified as potentially acceptable, and final Environmental Assessments were prepared for the five nominated sites (U.S. Department of Energy 1986b; U.S. Department of Energy 1986c; U.S. Department of Energy 1986d; U.S. Department of Energy 1986e; U.S. Department of Energy 1986f). Site Characterization Plans for the Deaf Smith and Hanford sites were nearly completed at the time work on those sites was terminated.

The U.S. Department of Energy granite program documents discussed earlier were prepared in the same timeframe as the Environmental Assessments. These documents identify seventeen states within which there were granitic bodies believed to be adequate for investigation for siting a repository for the second repository program (Figure A-3). The states identified included Minnesota, Wisconsin, Michigan, Maine, New Hampshire, Vermont, Massachusetts, Connecticut, Pennsylvania, New York, New Jersey, Delaware, Maryland, Virginia, North Carolina, South Carolina, and Georgia. Supporting references identify eight additional states under consideration by the crystalline rock program as having granitic bodies that could be adequate for investigation for siting a repository for the second repository program: Washington, Idaho, Arizona, Wyoming, Texas, Alabama, South Dakota, and Oklahoma (U.S. Department of Energy 2008a).

Of the original nine first repository program sites there remain eight sites that represent viable options. From the crystalline rock program, which included the second repository program, there are the original seventeen granite areas plus six additional granite areas (Texas and Washington are already included under the first repository program). Therefore, from the original first and second repository (crystalline rock) programs a total of thirty-one states have been identified that have potential sites or areas that could be evaluated for their potential for a second repository. These states are illustrated on Figure A-4.



Source: modified from U.S. Department of Energy 2008a.

Figure A-4. Areas and Sites in the Conterminous United States That Have Been Considered Previously for Repository Development

In order to increase the diversity of rock types under consideration by the geologic repository program, the U.S. Department of Energy initiated the Sedimentary Rock Program in 1984, the results of that study were not published until 2003 (Croff et al. 2003). The objective of that program was to evaluate five types of sedimentary rocks, including sandstone, shale, chalk, carbonate rocks, and anhydrock, to determine the potential for locating a geologic repository in one of these rock types. The report also included the results of a survey of foreign activities concerning sedimentary rocks other than salt that disclosed that only shale-like rocks were being seriously considered. At that time shales and/or clays (along with granite) were the geologic media of choice in Belgium, Italy, and Japan. Shales and clays were considered at that time to be alternatives in France, England, and Canada. Clays were also being considered in virtually every country for use as backfill or buffer material.

In that evaluation, shales were found to be equal to, or better than, the other four sedimentary rock types considered. Hard or rocklike shales having the favorable characteristics leading to this conclusion were seen to occur extensively in the conterminous United States. The outline of these shale deposits is included on Figure A-4. With the inclusion of these shale deposits, that figure shows that all states in the conterminous United States have an identified potential site or area that could have been considered for the second repository.

Appendix B

Analogue Information Relevant to Geologic Disposal

Appendix B—Analogue Information Relevant to Geologic Disposal

B-1. Introduction

This appendix presents information about natural and anthropogenic analogues for components and processes relevant to geologic disposal, and the sites where they are found, to provide insight into the long-term performance of these deep geologic disposal system components. This supplements the brief summary presented Section 2.1.2.4; Figure 2-1 illustrates the various geologic disposal facility components. At this stage of examining generic geologic disposal systems, there is little value in trying to develop site-specific analogue information. Rather, the following sections concentrate on the types of analogue information that will be available for building confidence as site-specific models are developed. Where generic media specific observations can be made, they are noted. Unless otherwise noted, the discussion follows Ruiz Lopez et al. (2004).

The study of these analogue sites provides information that is useful in developing geologic disposal facilities because it can enhance confidence in the projections of the long-term behavior of geologic disposal facility components. First, the analogue sites provide information about what types of materials are robust when subjected to the event and process phenomena that a geologic disposal facility would be expected to be exposed to over very long time periods. This aids scientists in justifying selection of media within which to develop the geologic disposal facility, and understanding those natural and engineered barrier components that are likely to provide robust performance. Next, the analogue sites provide data about the processes themselves, particularly how they evolve through time, lead to changes (or lack thereof) in the materials, and how changes in the materials lead to concomitant changes in the manner by which the event and process phenomena affect the materials. This leads to yet a third way in which geologic disposal facility analogue sites are of use. The very long timescales that must be considered in order to assure safe disposal of used nuclear fuel and high-level radioactive waste in deep geologic disposal facilities raise issues about confidence in the projections of performance. Analogues provide data to help build confidence in the safety assessment models and defend the long-timeframe performance projections that must be made with them to assess geologic disposal facility performance.

Section B-2 describes the features of a number of sites with analogue information relevant to geologic disposal. Section B-3 provides a summary of the process phenomena reflected in the models used in the safety assessments for which analogue information has been identified, and the sites where that information has been found. Section B-4 presents descriptions of the analogues of disposal materials and geologic disposal facility components, followed by descriptions of the analogues that illustrate the processes relevant to disposal system evolution.

B-2. Sites with Analogue Information Relevant to Geologic Disposal

The analogues at Pocos de Caldas, Koongarra, Cigar Lake, Oklo, El Berrocal, and Palmottu are uranium deposits. The main characteristics of and processes that take place in their surroundings that are of potential interest as analogues to a geologic disposal facility include the composition and long-term performance of uraninite, including stability, corrosion and dissolution, as an analogue of used fuel, the role of redox processes in radionuclide mobilization and retardation, including redox fronts and other geochemical discontinuities, the control of radionuclide speciation and solubility in groundwaters, including the formation and behavior of colloids, the retardation processes affecting the immobilized radionuclides, including the phenomena of sorption on surfaces and diffusion in the matrix, the possibility of using radioactive element series to estimate the longevity of various mobilization and retention processes, and the influence of colloids and microbial populations on radionuclide mobility.

The analogues at Kinnekulle and Dunarobba are concerned with the characterization of the long-term stability of bentonite. The main characteristics of and processes that take place that are of potential interest as analogues to a geologic disposal facility include the longevity and alteration of bentonite, the function of bentonite as hydrologic barrier and colloid filter, physiochemical changes (smectite to illite transformation) in bentonite caused by heating induced by the waste package, and the collapse of the waste package and interaction with other material of the engineered barriers. The longevity and degradation rate of bentonite (the smectite to illite transformation) is the most studied, both in analogues and scientific studies.

The analogue studies at Oman and Maqarin examined hyperalkaline environments. Hyperalkaline environments are natural occurrences of secondary minerals analogous to those formed during the hydration of Portland cement and result in interstitial waters characterized by very high pH. The study of this type of natural system may be of use in analyzing the safety of a geologic disposal facility, particularly with regard to the longevity of cement and its binding properties, the permeability of cement to water and gas, the speciation and solubility of radionuclides under high pH conditions, the interaction of high pH fluids with the surrounding rock, as an analogy to interstitial waters migrating from the disposal system to the host rock, the nature and viability of microbiologically induced geochemical processes and their influence on the processes of waste degradation, and the nature and stability of colloids formed in high pH waters and at the interface between these and neutral waters

At Maqarin, low-temperature reactions with the mineral phases of the natural environment, including hydration, carbonation, and sulfatization, have created a set of alternate alteration materials, many which are common to those existing in cement. The hyperalkaline upwellings at Oman have hydrochemical characteristics and associated mineral precipitates that are very similar to the conditions that are expected to exist in a disposal facility containing cement.

Archeological artifacts, including the Kronan cannon, the Inchtuthil nails, the Tournai sarcophagus, the water conduit at Acquarosa, Hadrian’s Wall, and the Changsha tomb can also provide analogue information. Archaeological analogues also provide information that is of value to demonstrating the safe long-term performance of repositories. The analogues investigated include the corrosion of cement or metal objects including iron, copper, and bronze analogous to waste containers or the waste materials themselves; the degradation of glass and cementitious or bituminous material as an analogue of the wastes; the long-term evolution of physicochemical properties of cements and other building materials analogous to the structure of the disposal system; the decay and breakdown products of organic material and complexation with trace elements, analogous to waste degradation; and chemical interactions between buried objects and host rocks or soils that might be analogous to near-field processes.

a. Oklo—The most distinctive characteristics of a number of the local uranium deposits at Oklo in southeast Gabon is the fact they were natural fission reactors in which nuclear chain reactions took place spontaneously about 2 billion years ago. Oklo is the only place known on earth where any significant deviation from the average value of the $^{235}U/^{238}U$ ratio occurs, indicating the activity of a natural reactor.

The natural reactors at Oklo are located in sedimentary basins; more than 15 reactors have been discovered, all located at depths of about 400 m, with the exception of one discovered at Bangombe at a depth of 11 m. In all cases the reactors are in the zone of contact between the formation through which oxidizing fluids rich in uranium flowed, and the organic matter rich formation where the uranium was accumulated in the form of uraninite as a result of the reducing conditions in the zone.

The reactors are zoned bodies characterized principally by two facies: (a) a reactor core, with a content of uranium of up to 87%, where the fission reaction took place; it is made up of uraninite grains embedded in a clay matrix; and (b) the reactor clays, which surround the core and are formed by hydrothermal clays, fundamentally chlorite, and illite. The different environments in which the reactors are located have made it possible to study and model their stability and the distribution of the radionuclides under different conditions and from different points of view.

The main objectives of the studies of Oklo were to understand the processes of uraninite alteration, and radionuclide transport and retardation over very long times.

The processes of alteration that occurred in these deposits make Oklo a unique analogue of the long-term behavior of fission and activation products in a high-activity waste deep geological disposal system.

The main contributions to safety assessments from the analogue studies at Oklo include:

- Support for the conceptual model of used fuel stability under reducing conditions; more than 90% of the uraninite in the reactors has remained stable for 2 billion years
- Support for the conceptual model of the isolation and retention capacity of the clay barrier
- Increased knowledge of radionuclide retention capacity in oxyhydroxides, iron, phosphates, and graphite, and of the low degree of influence of radiolysis due to buffering by organic matter
- Increased knowledge of criticality scenarios
- Confirmation of the control of redox conditions in the reactors due to radiolysis of the water and the presence of organic matter

b. Cigar Lake—The Cigar Lake uranium deposit is located in the Athabasca basin in the province of Saskatchewan, Canada. It was formed approximately 1.3 billion years ago by the activity of hydrothermal fluids, rich in uranium, at temperatures above 150° C. Situated at a depth of 450 m, there are no traces of its presence on the surface. It is a deposit of uraninite and pitchblende, surrounded by a clay halo, located in an active hydrogeological system with highly diluted waters that lose their initial oxidizing characteristics due to reaction with the host rock, becoming reducing. The studies performed on the system have focused on characterization of the processes of genesis of the deposits and on the processes that have led to the current distribution of radionuclides, taking into account their interaction with groundwater and all the processes that influence the transport and retention that might have affected them throughout their geological history.

Cigar Lake represents a large-scale analogue of high-activity waste disposal in crystalline formations. Its depth, at 450 m, is similar to the depths currently considered for mined geologic disposal. It is surrounded by a clay halo embedded in sandy minerals with active circulation of groundwater which is an analogue for a disposal facility’s bentonite barrier. The lack of any trace of it on the surface after its presence for millions of years is an analogue for the long-term isolation provided by a geologic disposal facility.

The main objectives of the studies of Cigar Lake were to understand the stability of uraninite, the study of transport through bentonite, and the role of colloids and organic compounds in radionuclide transport.

The main contributions to safety assessments from the analogue studies at Cigar Lake include:

- Support for the conceptual model of used fuel stability under reducing conditions
- Support for the conceptual model of geohydrologic isolation due to filtering of colloids by the bentonite buffer, represented by the clay halo
- Development of sophisticated radiolysis models
- Support for the conceptual model of irreversible nuclide sorption of colloids
- Quantitative data on the solubility of trace elements
- Quantitative data on uraninite dissolution rates of 10^{-8} to 10^{-9} per year

c. Palmottu—The Palmottu uranium and thorium mineralization is located in southwest Finland and is associated with gneiss and crystalline granite rocks. This mineralization was formed as a result of hydrothermal activity nearly 1.8 billion years ago. The ore forms a vertical structure parallel to a granitic dyke, which extends to depths of about 400 m, and is calculated to contain 1,000,000 tons of ore.

During recent geological history the area has been subjected to glaciation, the last episode ending about 10,000 years ago. Glaciation alternated with interglacial periods, allowing the infiltration of oxidizing waters toward deep zones from ice melting and moving through fracture systems. Interaction between the rock and the groundwater has led to a series of characteristic hydrochemical subsystems that have conditioned the mobilization of uranium and thorium. These elements, initially contained in the mineralization, have migrated toward the fracture zones; close to the fracture zones, secondary minerals have been reprecipitated.

The main analogues found in the system are the presence of uraninite, as an analogue to used fuel, subjected to different alteration processes and the effects of these processes on radionuclide mobilization and retention in a crystalline rock similar to those considered for geologic disposal systems

The main objective of the study of the analogue at Palmottu was to better understand radionuclide transport in fractured crystalline rock masses.

The main contributions to safety assessments from the analogue studies at Palmottu include:

- Support for the conceptual model of used fuel stability considering the dissolution of uraninite
- Confirmation of the ability of the rock to maintain reducing conditions despite the infiltration of oxidizing water
- Support for the conceptual model of radionuclide retention by precipitation, coprecipitation, and sorption of radionuclides on minerals filling fractures
- Quantitative data on the penetration depth by matrix diffusion, which was limited to 25 mm
- Qualitative data on paleohydrologic aspects of glaciation

d. Shinkolobwe—Shinkolobwe is a uranium deposit ore body in Zaire. Comprehensive investigation of the corrosion products of uraninite was undertaken on the material from the Shinkolobwe ore body (Miller et al. 1994; Miller et al. 2000). The Shinkolobwe deposit weathers under oxidizing conditions in a monsoonal type environment where rainfall is about 1000 mm per year. At Shinkolobwe, the uraninite is coarsely crystalline and lacks many of the impurities, for example thorium and rare earth elements, found in most uranium deposits. This lack of impurities has led to the suggestion that the thermodynamic stability of the Shinkolobwe uraninite may closely approximate that of used fuel. Many secondary or secondary uranyl phases were identified from the alteration of uraninite at Shinkolobwe. The conditions at Shinkolobwe are very different from the reducing environment expected in a repository near field. As such, the Shinkolobwe natural analogue might be considered most relevant to oxidizing conditions persisting in the near field, which typically would not be an expected condition for a deep geologic disposal facility. The Shinkolobwe ore body in an oxidizing environment would be somewhat representative of a near field made oxidizing by the buildup of radiolytic oxidants.

It has been suggested that the natural occurrence of some uranium minerals may indicate the existence of highly oxidizing conditions resulting from radiolysis (chemical decomposition by the action of radiation). The identification of radiolysis at Oklo, Cigar Lake, including Cluff Lake and Rabbit Lake, and Shinkolobwe, which collectively represent a range of geological environment and chemical conditions, suggest that radiolysis is a common feature in nature in systems where naturally high radiation fields occur. It is likely that the radiolytic processes that would occur in a repository would be identical in mechanism to those observable in natural systems. However, the rates of radiolysis might be different due to the different radiation fields in a used nuclear fuel or high-level radioactive waste repository as compared to a uranium ore body.

The main objective of the study of the analogue at Shinkolobwe was to better understand the dissolution processes uraninite would undergo under oxidizing conditions.

The main contributions to safety assessments from the analogue studies at Shinkolobwe include:

- Evidence suggests that radiolysis may lead to near-field oxidizing conditions, which should be examined in safety assessments.
- The lack of impurities in the uraninite has led to the suggestion that the thermodynamic stability of the Shinkolobwe uraninite may closely approximate that of used fuel

e. The Kronan Cannon—The Kronan was a Swedish warship built in 1668. During a battle the warship was sunk by the German-French fleet in the Baltic Sea a distance of 5 km from the coast. Between 1680 and 1686 approximately 60 cannons were recovered from the seabed. More recently, in the 1980s, another 32 cannons were recovered.

The subject of this analogue study has been one of the bronze cannons, which remained vertically positioned, with the muzzle downward, with 1.6 m of its length buried in the seabed and the remaining length above the sea bottom in direct contact with the water. The Kronan cannon has a very high copper content, nearly 97%.

The marine clays in which the cannon was buried are similar enough to be considered an analogue of bentonite. The Kronan cannon provides information on the type of corrosion the copper undergoes in a clay matrix with oxidizing properties. The average corrosion rate of the copper was determined to be 0.15 μ (0.00015 mm) per year.

The corrosion of a copper canister in a geologic disposal facility likely would be less in the reducing conditions typically expected to exist in a geologic disposal system.

The Kronan cannon provides evidence of the corrosion resistance and longevity of copper; this information might be used to increase confidence in the suitability of copper as canister material.

f. The Tournai Sarcophagus—At Tournai, in Belgium, a French Roman lead sarcophagus from about 300 A.D. was discovered in 1989. The sarcophagus is made up of two parts; a tub and a cover. The tub is formed by a sheet of lead measuring from 6 to 9 mm in thickness, folded in the shape of a trunk. The wall joints are made by lead-zinc welding. The cover was manufactured in the same way. Buried at the end of the third century, the lead sarcophagus contains the body of a man, a glass, a cup, and about 50 pieces of bronze. Within the sandy clay environment in which it was found, the sarcophagus has been subjected to hydrostatic pressures that have caused sediments to penetrate the top. From the point of view of the mechanical performance of the container, the rapid filling of the sarcophagus gave it greater resistance to deformation. Chemically, the sarcophagus is made up of metallic lead, protected by a layer of corrosion products including lead oxides, lead carbonates, and lead phosphates. The corrosion products form a fairly homogeneous protective shield; this has preserved the metallic lead against subsequent attacks.

The study of the Tournai sarcophagus has allowed insight to be gained into the corrosion processes that affect the parts of the metal buried in a relatively permeable, aerated, and periodically saturated geological environment.

While lead typically is not used as a protective covering for high-level radioactive waste disposal canisters, the study of the Tournai sarcophagus demonstrates the fact that the corrosion products of a metallic container may contribute to maintaining the integrity of a container over long periods of time.

g. The Inchtuthil Nails—The Roman legion fortress at Inchtuthil is located in Scotland and was abandoned in 87 A.D. During their withdrawal, the Romans placed about 875,000 nails in a hole measuring 5 m in depth, which they buried by filling it with 3 m of compacted earth. The nails remained buried until 1950 when the fortress was excavated and the nails recovered.

The nails exhibit heterogeneity in their composition, with many variations in carbon content. The degree of corrosion of the nails depended, among other factors, on their position the pile. The nails located in the interior of the pile show minimal corrosion, while for those on the outside, and in particular those on the top of the pile, the corrosion was so intense that a large layer of iron oxide formed. In certain of the nails in very limited areas, localized pitting corrosion was observed; it might be that the pitting was influenced by the composition of the iron.

At Inchtuthil the corrosion rate of the outer nails was larger because they were subjected to oxidizing conditions. These nails acted as a barrier against the corrosion of those located in the interior of the pile.

In general terms the situation may be considered analogous to the one to which the steel waste packages and other engineered components will be subjected in a deep geological disposal system. Qualitatively, it may be concluded that where there are large volumes of steel in a repository, a large part of it might not be affected by corrosion. In this analogue example, it has not been possible to identify the additional information, for example pH, Eh, and groundwater chemistry required for quantitative valuation of all the variables involved.

h. Kinnekulle—At Kinnekulle, in Sweden, about 450 million years ago, volcanic ash was deposited on a series of clay and calcareous sediments. The ash deposits were subsequently covered by layers of marine sediments measuring several hundred meters in thickness. The volcanic ash was transformed into bentonitic and smectitic layers that were subsequently consolidated by the vertical pressure of the overburden materials. Approximately 300 million years ago, the rock layers were intersected by a magmatic intrusion, tens of meters thick. The thermal pulse associated with this intrusion affected the surrounding sediments and contributed to their transition to consolidated sedimentary rock.

Indirect methods of investigation indicate that the temperature to which the layers were subjected did not exceed 160° C. The degree of alteration of the bentonite as a result of this heating, including both the transformation of smectite into illite and cementation by silica, was also studied.

The objectives of the study of the analogues at Kinnekulle were to analyze the processes that lead to cementation of smectite affected by heat.

The Kinnekulle bentonites are a good analogue to the performance of the clay barrier in a geologic disposal facility, considering the longevity of the barrier and the physiochemical changes due to heating.

The principal contributions made by the analogues at Kinnekulle to safety assessments are:

- Support for the conceptual model of clay barrier performance despite the chemical effects on the bentonite; the degradation suffered by the bentonite at Kinnekulle has not caused any alteration of its properties as a barrier to radionuclide release
- Identification through sensitivity analysis of the parameters in the process of the bentonite transformation, including, temperature, concentration of potassium, and activation energy

i. Oman—The analogue studies at Oman examined five hyperalkaline upwellings found near Muscat in northeastern Oman. The upwellings occurred in a mountain chain that parallels the coast and comprises ocean floor igneous rocks. During the uplift of these materials, surface waters penetrated them and subsequently, on moving downward and interacting with the ultramafic minerals of the rock, became transformed into hyperalkaline, and frequently very reducing, groundwaters with pH values around 11. These waters later returned to the surface resulting in the hyperalkaline upwelling observed.

Studies at two of the upwellings show a wide variety of microbial populations present in the water that were adapted to the extreme conditions. The geochemical modeling carried out on these waters focused on the applicability of geochemical codes and thermodynamic databases used in safety assessments. The models previously had not been verified for systems having characteristics as extreme as these hyperalkaline deposits under reducing conditions.

Disposal design concepts currently being considered may include large volumes of concrete that can interact with, and condition, the waters in the near field, resulting in very high pH values. The waters at Oman are very similar to these, since they are strongly alkaline and reducing. For this for the reason

Oman constitutes an analogue for the study of such hyperalkaline waters and their effects on the process of radionuclide transport in the disposal system.

The main contributions made to safety assessments from the analogue studies at Oman include support for models of radionuclide release and migration in a geochemical environment that is difficult to simulate in the laboratory, checking the adaptation of microbial communities to extreme conditions, confirmation of the degree of influence of colloids in spite of their abundance due to instability under these conditions, and verification of the applicability of redox models.

j. Pocos De Caldas—Pocos de Caldas is a volcanic caldera, 35 km in diameter, located in southern Brazil in one of the areas of the highest natural radioactivity levels in the world. The caldera was formed 75 million years ago and subsequent hydrothermal activity in the area led to the formation of numerous metallic deposits that are worked with open pit mines, principally producing uranium metal along with lesser amounts of thorium and rare earth elements. The most important characteristic at the mine is the formation of a redox front due to the remobilization of uranium and other elements from the mineralized zone. This is associated with the appearance of uranium enrichment on either side of the front, as pitchblende in the reduced zone, and associated oxyhydroxides of iron in the oxidized zone.

The front is due to the infiltration of oxidizing storm waters and their interaction with the rocks. The oxidized and reduced zones correspond to past conditions in which weathering penetrated preferentially along fractures. At present, the hydrogeology is highly influenced by the mining of the deposit causing an ascending flow of reducing waters in what previously was an oxidizing water recharge area.

The principal objective of the studies at Pocos de Caldas was the geochemical characterization of the transport of radionuclides sensitive to changes in redox conditions.

The main analogues are as follows:

- The dissolution of uraninite and pitchblende is analogous to the process of dissolution of used fuel
- The retention processes of uranium and other trace elements in the iron oxyhydroxides in the oxidized area of the redox front are analogous to the possible retention processes in the corrosion products of the canister
- The redox front developed is analogous to the zone that might be developed in the near field of the disposal system as a result of radiolysis
- Characterization of the interactions between the radionuclides, colloidal phases, and mineral surfaces helps to understand colloidal stability and the processes of migration of these elements in a natural system

The main contributions to safety assessments from the analogue studies at Pocos de Caldas include

- Support for the conceptual model of used fuel stability under reducing conditions
- Support for the conceptual model of canister corrosion: the radionuclides are immobilized by co-precipitation and sorption in the oxyhydroxides of iron
- Quantitative data on the rate of advance the redox front: 1 to 20 mm in 1000 years
- Checking different approaches to reactive transport modeling
- Verification of thermodynamic databases under different redox conditions

k. Maqarin—Maqarin, in northwest Jordan is characterized by the development of a layer of cement, due to the alteration of metamorphic minerals, generated during the spontaneous combustion of marls that were very rich in organic matter; temperatures of up to 1000°C were reached. The subsequent interaction of seepage waters with the natural cement minerals led to the development of groundwater with very high pH values, oxidizing conditions, and a set of secondary minerals characteristic of the evolution of a hyperalkaline aureole.

In addition to the characterization of the solid materials in the cement zone, a study was carried out on the groundwaters and their interaction with the host formation. The study allowed the development of a conceptual model of the evolution of a hyperalkaline aureole. Also studied was the distribution of trace elements in the cement minerals and the secondary minerals generated by interaction with a hyperalkaline plume.

Maqarin is the best-known analogue of the long-term behavior of a hyperalkaline aureole expected to be developed in a disposal system containing cement. In the case of the hyperalkaline waters of Maqarin and the waters of a hypothetical disposal system including cement, there are three different stages of evolution: 1) percolation of rainwater; 2) interaction between these waters in the cement; and, 3) flow of hyperalkaline waters containing elements captured during the processes of interaction with a fractured rock and the rock matrix. The main information provided by this analogue relates to the behavior of certain trace elements under extreme geochemical conditions, as well as to the evolution of the characteristics of the water.

At Maqarin an unusual assemblage of secondary minerals, which are the result of interaction between the hyper alkaline groundwater and the rock, has been observed. As part of the Maqarin project these analogue observations were used to help develop and constrain a conceptual model to explain the possible interactions that might occur from a hyperalkaline plume migrating away from a geologic disposal facility through the host rock. The model assumes that cement leachate rich in sodium, potassium, and calcium flows outward from the repository driven by the groundwater flow system. As the plume begins to interact with the host rock, a complex sequence of reactions involving dissolution of the aluminosilicate minerals in the rock and precipitation of calcium silicate hydrate compounds can occur. As the pH decreases and the aluminum concentration increases, eventually zeolites can form (Miller et al. 2000).

The main areas of interest for this analogue study were investigation of the overall hyperalkaline groundwater evolution, including the question of the evolution of the cement leachate, interaction of the hyperalkaline leachate with the host rock, and the testing of a variety of geochemical transport and biological codes. The hydrogeology of the Maqarin site is complex and at least two geochemically distinct flow systems exist. In the eastern part of the area, the groundwater is pH 12.5 and is buffered by abundant Portlandite in the source rock. In the western part of the site the groundwater contains much higher levels of sodium and potassium and appears to be a younger system (Miller et al. 2000).

The main contributions made by the analogue at Maqarin to safety assessments include:

- Support for conceptual models and development of new models of the evolution of hyperalkaline aureoles in a disposal system containing cement, and their orientation with respect to groundwater flow
- Evaluation of the delay in the transport of non-sorbing radionuclides transported by advection by the hyperalkaline waters through fractures that are open
- Support for the applicability of geochemical and reactive transport codes and thermodynamic databases under extreme conditions

l. El Berrocal—El Berrocal, in Spain, is located about 92 km south of Madrid in the north-central part of the country. It is a granitic pluton, enriched in uranium and thorium, with an outcropping surface area of 22 km². The emplacement of the pluton is estimated to have occurred nearly 300 million years ago.

The geological environment of El Berrocal is characterized by intensive fracturing, which favored hydrothermal activity in the areas close to the fractures at least 1 million years ago. As a result, there was remobilization of uranium, thorium, and other elements initially contained in the pluton in the form of dispersed uraninite, and the subsequent reprecipitation in a quartz and uranium mineralized dyke of secondary mineralized veins.

The subsequent weathering processes remobilized the uranium and thorium and other elements which were subsequently retained by sorption or precipitation in the silicate, phosphate, carbonate, clay, and oxyhydroxide minerals filling the fractures. The hydrogeological system is strongly conditioned by the discontinuities present, including fractures and dykes of quartz and uranium, which constitute the preferential pathways for groundwater flow.

The main objective of the studies at El Berrocal was to gain understanding of natural radionuclide transport processes in a fractured granite environment.

The main analogues encountered in this system are the presence of uraninite and pitchblende as analogues of used nuclear fuel, subjected to different alteration processes, and the effects of those processes on radionuclide mobilization and retention both in the fracture filling materials and as a result of diffusion in the matrix.

The main contributions to safety assessments made by the analogues at El Berrocal are as follows:

- Support for the conceptual model of used fuel stability
- Support for the conceptual model of matrix diffusion and retention in the geosphere by means of the processes of precipitation, coprecipitation, and sorption on fracture filling minerals
- Development of models of radionuclide transport affected by fluoridated and carbonated complexes and microbial activity

m. The Acquarosa Conduit—In the Viterbe region of Italy, at Acquarosa, there is an underground water conduit dated at between the sixth and seventh centuries B.C. The floor and walls of this conduit are coated with concrete that is in a perfect state of preservation. The conduit is a tunnel measuring 80 cm in width and approximately 1.8 m in height. The base of the channel measures 15 cm in thickness while the average thickness of the walls is 5 cm. The upper part of the channel is vaulted with a height of 35 cm. The tiles on the floor consist of bricks joined with a carbonated gypsum. The surface of this area is rough, while the walls are of the same type of concrete but finer and have a smoother finish. The upper parts of the walls are covered with a layer of carbonated plaster with a highly polished surface. The plaster has not altered significantly with time; the calcium silicates found probably came from the reaction between the carbonates and the sand or brick dust present. The most important analogue at Acquarosa is the good degree of preservation of concrete containing carbonates inside silicate rocks.

n. Hadrian's Wall—In 122 A.D., the Emperor Hadrian ordered the construction of a wall crossing Great Britain from Solway Firth in Scotland to Wallsend in England. The wall measured 117 km in length at an average of 5 m in height. The wall was made of blocks of stone with a cement that acted as a binder. Small fortifications, watchtowers, and fortresses were built along the wall. After an uprising in 1745, a part of the wall was destroyed, and today the best preserved parts measure only 1 m in height. Hadrian's Wall contains significant quantities of the hydrated calcium silicate compounds that are the basis of modern Portland cements. The mortar was compacted and has a low porosity. Practically all the calcium oxide in the mortar was carbonated, which along with the evidence of reaction in certain aggregates, confirms the presence of hydrated calcium silicate compounds.

Hadrian's Wall is of interest as an archaeological analogue due to the excellent state of preservation of the cement used to join the stone blocks. The surface environment in the north of England is very different from the conditions that could be found in a geologic disposal facility; however the chemical and mineralogical similarities between the cement used by the Romans and modern Portland cement allows qualitative conclusions to be drawn regarding stability and longevity, which might be extrapolated to the modern cements used in a repository.

The main contribution of the Hadrian's Wall concrete to safety assessments is its illustration of the durability of the cement that might be used in engineered barriers in a geologic disposal system.

o. The Tank at Uppsala Castle—A study was performed on an old water tank that had been installed in the towers of Uppsala Castle in Sweden (Miller et al. 2000). The tank was installed in 1906 and was demolished in 1991. The steel tank was lined with a 20 mm-thick layer of cement mortar. In the 85 years the tank was in operation, the tank was regularly refilled with fresh water. The mortar was continuously being leached because equilibrium could never be reached between the cement and the water in the time available. The concrete mortar was investigated by chemical, physical, and optical methods to understand the performance of the concrete lining in this environment.

The results showed that the mortar was covered with a thin layer of carbonates that were believed to have formed by reaction between the bicarbonate in the water and the cement. Under the bicarbonate layer was a 5 to 8 mm thick zone of enhanced porosity, reduced calcium content, and a relatively increased sulfate and iron content. This represents a region of complex leaching, redistribution of elements, and re-crystallization. Portlandite was not apparently depleted in the porous zone, although it had recrystallized to coarser aggregates and calcium silicate hydrate compounds, and reorganized to a lower calcium silica ratio. Cement leaching can be modeled in safety assessments with a model that assumes instantaneous release of leachates to the water. Applying such a law to the water tank leads to a prediction that the leaching depth should be 6 cm and that all Portlandite should have been dissolved.

The main contribution to safety assessments from the analogue studies of the Tank at Uppsala Castle is that the analogue data clearly indicate that diffusion controlled leaching proceeds at a slower pace than predicted by the performance assessment model, and thus the model is conservative.

While 85 years is a much longer time than can be studied in a laboratory, it is still very short compared to geologic disposal facility lifetimes. As such the water tank analogue falls short of being able to validate the rate of progressive decrease in pH predicted in the safety assessment models. One possible way to circumvent this problem is to examine the evolution of groundwater which is naturally highly alkaline. The natural analogue most suitable to constrain hyper alkaline groundwater evolution and interaction with the host rock is Maqarin in a northern Jordan, described previously.

p. The Changsha Tomb—In Changsha, Hunan province, southern China, a burial site located in a rice paddy has been found. The body was buried at a depth of 16 m. Burial sites as elaborate as the Changsha tomb were not common in the Chu and Han dynasties between 770 B.C. and 220 A.D. The tomb measures 16 m in depth and was located in a soil for which the characteristics were not reported in the literature. A layer of natural coal measuring 30 to 40 cm in thickness was placed in contact with sarcophagi to absorb the humidity of the soil. Around this coal layer there was another layer of kaolinite, which was what gave the system its waterproof characteristics. The outer sarcophagus contained a large number of funeral offerings, including silk garments, bamboo objects, musical instruments, bronze artifacts, and even food. The sarcophagus in the innermost part of the shaft contained the body, carefully wrapped in different cloths, including silk and linen. A study of the body revealed that it was in an exceptional state of preservation. The fundamental reason for the state of preservation of the body and of the materials found in the sarcophagus is the layer of kaolinite that prevented water and air from seeping into the sarcophagus. There is a direct relationship between the thickness of the layer of kaolinite and its effectiveness in preserving the body. The Changsha tomb illustrates the isolating capacity of clay. This type of burial constitutes an analogue of the clay barrier of the multiple barrier system contemplated in most radioactive waste disposal repository concepts.

The main contribution to safety assessments from the analogue studies at Changsha Tomb is the evidence that it provides of the isolating and sealing capacity of the clay layer that would be expected to be present in a radioactive waste disposal system.

q. Koongarra—The uranium deposits of Koongarra are located in a geosyncline at Pine Creek in northern Australia, in the region of the Alligator Rivers. The deposits were formed about 1.8 billion years ago as a result of significant hydrothermal activity. Although the area includes various mineralized bodies rich in uranium, the Koongarra deposit is the one that has undergone the greatest alteration and weathering over the last several million years.

The majority of the groundwater flow takes place in a zone through partially weathered schists of the host formation. Recharge to this aquifer occurs via a fault and by direct infiltration of rainwater. This recharge leads to dilution and vertical stratification of the hydrochemical characteristics of the formation. Given the intense weathering, the uranium and other elements initially contained in the mineralized veins in the form of uraninite and pitchblende have undergone dissolution and subsequent precipitation, moving into the groundwater and being retained by different processes. These processes gave rise to different zones of secondary mineralization and a uranium dispersion halo about 80 m from the mineralized body.

The principal objective of the studies at Koongarra was to support the development of radionuclide transport models.

The effects produced by uraninite leaching under oxidizing conditions, radionuclide migration and retention, and the movement of weathering fronts are excellent analogues to the behavior to be expected in a high-activity waste geologic disposal facility in an advanced phase of degradation under normal conditions or in the event of loss of engineered barriers and sealing capacity.

The main contributions to safety assessments from the analogue studies at Koongarra include:

- Support for the conceptual model of used fuel stability under reducing conditions
- Support for the conceptual model of immobilization by precipitation, coprecipitation, and sorption of radionuclides on secondary minerals
- Development of conceptual sorption models
- Confirmation of the low degree of relevance of matrix diffusion
- Confirmation of the importance of the alpha recoil process
- Quantitative data on the migration rate of uranium: 25 to 100 mm in 1000 years

r. Dunarobba—The fossil forest at Dunarobba, located in central Italy, consists of 50 trunks of fossilized trees that are still in their original position dating back approximately 2 million years. An outstanding characteristic of these fossil trees, that makes them different from others existing in the geological record, is that they remained wood after being buried. The Dunarobba forest developed 2 million years ago in a marshland beside a rather shallow lake that filled with pelitic (argillaceous) materials, including clays and silts laminated with frequent beds of sand and gravel and abundant organic matter. Given the high rate of sedimentation, of about 3 m in 1000 years, the forest was buried and rapidly isolated from oxidizing conditions.

Above the level at which the forest is located, the stratigraphy becomes more homogeneous and is made up almost completely of laminated silty clays typical of a marshland environments; they are much less permeable than the underlying clays. The degree of preservation of the trunks is due to the isolation generated by these materials, which acted as a geological barrier to the processes of organic degradation. The area generally has been affected by active neo-tectonic processes and the gas distribution the soil also indicates that is located on one of the preferential routes for fluid migration in the basin. These processes would not be expected to favor the preservation the fossil forest.

At basin scale the pelitic material does not behave as an effective barrier against the migration of fluids due to short circuits caused by fractures. At a smaller scale, however, the fractures form the boundaries for blocks that behave coherently as a barrier due to their low hydraulic conductivity. The forest is located in one of these blocks, which would explain its preservation in an area in which there is a system of discontinuities in the materials.

The principal objective of the studies at Dunarobba was the analysis of the isolation capacity of clay materials.

The fossil forest at Dunarobba is considered to be an analogue of the long-term isolation capacity of clay materials, since these have isolated and preserved organic matter over a timescale of millions of years.

The main contribution to safety assessments made by the analogue at Dunarobba is support for the conceptual model of the performance of bentonite as a hydrologic barrier and as in limiting microbial degradation.

s. Loch Lomond—Loch Lomond is located in central Scotland; perhaps the best-known natural analogue study of radionuclide transport within sediments was performed there (Miller et al. 1994). Through the examination of a few sediment cores from the loch, valuable quantitative information has been gained. At present Loch Lomond is freshwater and landlocked; about 6000 years ago a marine transgression from the Firth of Forth resulted in an incursion of seawater into the loch. The event is clearly recorded in the sediments as a 1 m thick band of marine deposits that are both overlain and underlain by freshwater sediments. Consequently there is a geochemical discontinuity. The sediments are clay rich containing about 80% clay in some horizons. The presence of higher concentrations of chlorine, bromine, and iodine also serve to delineate the marine sediment horizon. The migration of salts from the marine sediments into the freshwater sediments and pore waters above and below the marine sediments records a history of diffusive transport. No mobile redox front was established in the Loch Lomond sediments (Miller et al. 2000). The pore water concentrations of bromine, and to a lesser extent iodine, decrease with distance from the marine sediments. The bromide concentration profile was modeled using simple diffusion with reversible sorption assuming no component of advective transport. The model produced an apparent diffusivity of 8×10^{-11} m^2/s. Laboratory batch experiments resulted in measured diffusivities an order of magnitude less than that modeled; this was attributed to sample disturbance. The laboratory experiments also indicated that some sorption processes are irreversible.

The principal aim of the Loch Lomond natural analogue study was to investigate diffusion and retardation mechanisms and rates of elemental transport that occur in argillaceous sediments as an analogue for transport and retardation processes in clay buffer materials or sedimentary host rocks.

The main contributions to safety assessments made by the study at Loch Lomond relate to modeling of diffusive radionuclide transport in argillaceous sediments, as an analogue to the behavior of clay / bentonite buffers or backfill. A principal conclusion of the study was that modeling of the natural situation in comparison to laboratory studies suggested that the calculated diffusivities were probably more realistic than the laboratory studies. Also, the assumption made in many models of migration that sorption is instantaneous and reversible is likely to be inaccurate.

t. Tono Mine—The Tono region, located 330 km southwest of Tokyo is the site of Japan's most extensive uranium deposits. The ore bodies at Tono lie in channels in the unconformity between Cretaceous granitic basement rocks and overlying Miocene lacustrine sediments, which themselves form the lowest unit in a 200-m thick pile of marine and lacustrine sediments (Miller et al. 1994). The basement granite rocks contain about 6 ppm uranium and are considered to be the source of the uranium mineralization. The uranium mineralization occurs in conglomerate, sandstone, and lignite formations. The ore body is thought to have formed when oxidizing groundwater leached uranium from the granite and transported it upwards to the lignite bearing rocks where the uranium was precipitated or absorbed or both under the more reducing conditions that prevailed there. The uranium concentration process continued for a long period of time about 10 million years ago. No substantial recent mobilization of uranium has occurred since this time despite later uplift erosion and faulting in the area.

The objective of the natural analogue studies at Tono was to evaluate the process of uranium series radionuclide migration, over timescale of about 10 million years, in both the pristine ore and in the ore around the faults.

The main contributions to safety assessments are related to demonstration of limited movement of radionuclides. Samples from the mine and nearby boreholes were analyzed in uranium series disequilibrium studies. The results indicated that the reducing conditions have been maintained for at least the last ten million years and that although limited uranium migration has occurred along the faults, the greatest uranium migration has occurred in the matrix of the ore. This has been limited to less than 1 m in the last million years.

B-3. Associations of Analogue Sites and Physical Phenomena

Table B-1 provides a summary of those physical process phenomena reflected in the models comprising the safety assessment simulation for which analogue information has been identified, and the sites where that information has been found.

Table B-1. Associations of Analogue Sites and Physical Phenomena

Physical Phenomena	Analogue Sites
Alteration of Used UO_2 Fuel Matrix	Oklo: reducing conditions Cigar Lake: reducing conditions Palmottu: reducing conditions Shinkolobwe: oxidizing conditions
Criticality	Oklo
Radiolysis	Cigar Lake Shinkolobwe
Corrosion	Kronan Cannon Tournai Sarcophagus The Inchtuthil Nails The Changsha Tomb
Dissolution – Precipitation in Bentonite	Cigar lake
Cementation in Bentonite Layer	Kinnekulle
Smectite-to-Illite Transformation	Kinnekulle Cigar Lake
Solubility Limits	Oman Pocos de Caldas Cigar Lake Maqarin El Berrocal Oklo Palmottu
Cement Degradation	Acquarosa Conduit Hadrian's Wall The Tank at Uppsala Castle Oman Maqarin
Fluid Flow	Oklo Cigar Lake Palmottu El Berrocal Pocos de Caldas

Table B-1. Associations of Analogue Sites and Physical Phenomena (continued)

Physical Phenomena	Analogue Sites
Advection and Dispersion	Oklo Palmottu El Berrocal Maqarin
Diffusion	Cigar Lake Loch Lomond Koongarra Dunarobba Palmottu Pocos de Caldas El Berrocal
Water Rock Interactions	Cigar Lake Pocos de Caldas Palmottu El Berrocal
Retardation Processes	Loch Lomond Palmottu El Berrocal Oklo Koongarra Pocos de Caldas Tono Mine Maqarin
Redox Front	Palmottu Oklo Pocos de Caldas Cigar Lake
Colloid Generation and Transport	Cigar Lake Palmottu Pocos de Caldas Koongarra El Berrocal Maqarin
Microbial Processes	Palmottu Pocos de Caldas El Berrocal Oman Maqarin

B-4. The Value of Analogue Evidence for the Strength of Geologic Disposal

B-4.1 Analogues of Disposal Materials and Components

Natural analogues represent the occurrence of materials similar to those found in a geologic disposal facility. The principal disposal materials and components of a deep geologic disposal system include the chemical forms of the of the radioactive waste, that is, the used nuclear fuel and vitrified high-level waste, the metal canister, including iron and copper materials, the backfill, buffer, and sealing materials, (e.g., bentonite, and cement and concrete), and the geosphere, the geological medium within which the deep geologic disposal system is developed. In the following sections, information about each of these materials is presented to set support discussions on evidence of why there can be confidence in projections of the long-term behavior of a geologic disposal facility and its components when subjected to expected repository processes.

B-4.1.1 Radioactive Waste: Chemical Forms

High-activity wastes consist of used nuclear fuel and high-level radioactive waste generated from the reprocessing or recycling of nuclear fuel. Disposal of used nuclear fuel without reprocessing is known as an open fuel cycle. In contrast, a closed fuel cycle involves reprocessing of the used nuclear fuel and disposal of the remaining high-level radioactive wastes, (under current practices the high-level wastes typically are immobilized in borosilicate glass, although other forms are possible).

There are significant amounts of high-level radioactive wastes in the U.S. both from defense related activities as well as from a past attempt at commercial used nuclear fuel reprocessing.

B-4.1.1.1 Used Nuclear Fuel

Used nuclear fuel comprises principally ceramic-like uranium dioxide (UO_2) normally with more than 95% UO_2. This form was chosen because of its great stability at high temperatures and its low solubility. In a geologic disposal system, this high degree of stability significantly limits radionuclide dissolution in the event of degradation of the engineered barriers, resulting in the used nuclear fuel itself being a principal barrier in the geologic disposal system.

The natural uranium oxides that appear in deposits of uraninite and pitchblende are considered analogues to used nuclear fuel and provide information on long-term behavior that complements that obtained from short duration experiments. These minerals have the UO_2 composition of used nuclear fuel; from a crystallographic perspective, they are identical.

The study of uraninite and its surrounding environment provide important information for predicting the long-term behavior of the used UO_2 fuel matrix, particularly as it pertains to the processes of dissolution and release of radionuclides from the matrix, and their retention in secondary alteration products.

B-4.1.1.2 Vitrified High-Level Waste

The principal high-level radioactive waste form resulting from the current practice of vitrification of the byproducts of reprocessing used nuclear fuel is borosilicate glass. Borosilicate glasses exhibit physical and chemical longevity under expected geologic disposal facility conditions. The natural analogue materials corresponding to the glass matrices are volcanic glasses, which are primarily aluminum silicates in composition. Among these the basaltic glasses are the natural materials most similar to borosilicate glasses. However, the natural glasses do not contain the radioactive components found in high-level radioactive waste, so it is not possible to observe the radiation-induced effects that might occur in a geologic disposal environment.

The study of archaeological glasses may help to understand the shorter-term behavior of the vitrified waste as well. Many ancient glasses have been subjected to temperature and humidity conditions that may be more unfavorable than those to be expected in a geologic disposal system without major alteration occurring. There are abundant studies on natural glass analogues, although these have been carried out from a geological or mineralogical point of view rather than for direct use in building confidence in safety assessments. Nevertheless this allowed the development of data on the behavior characteristics of these glasses.

Glass tends to degrade or change with time. Most natural glasses have ages of more than 2 million years, and the glass durability has been observed to be greater in those cases in which these materials have been prevented from coming into contact with water. It has been observed that low temperatures and very low flow rates considerably limit the hydration of the glass, making its dissolution a very slow process. Anthropogenic or archaeological glasses appear to degrade as a result of the same mechanisms observed in laboratory experiments on borosilicate classes.

B-4.1.2 Metal Canister: Iron and Copper Materials

Metals are used in the engineered barrier system of a geologic disposal facility for waste packages and canisters containing the used nuclear fuel or high-level radioactive waste. They also can be used as reinforcement in the concrete structures supporting the excavations. It is not unusual for most of these components to be made of steel; however, other metals such as copper in the Swedish and Finnish programs and titanium in the Canadian program have been considered for use for the waste packages. Furthermore metals may be present as components of the wastes themselves, as in the case of used nuclear fuel assemblies.

In most geological disposal concepts the waste packages are designed to serve as a barrier to contain and isolate the wastes for a period of several thousand years, as well as to provide resistance against lithostatic pressure and the stresses arising from swelling of any backfill or buffer material and the increase in volume associated with the corrosion products formed as the packages degrade. The use of thick metal waste packages also reduces the importance of radiolysis processes (chemical decomposition by the action of radiation) and can introduce significant amounts of iron into the system, contributing to the establishment and maintenance of a reducing environment (no free oxygen) around the waste package and waste form which enhances isolation characteristics. Many short-term laboratory experiments have provided abundant information on the degradation of waste package materials by corrosion processes. These results are not easily extrapolated to periods as long as that needed to study geologic disposal systems. Anthropogenic or archeological analogue studies have provided information on these effects covering scales of hundreds to thousands of years, helping to build confidence in the needed extrapolations.

A main difficulty in identifying analogues for the metals used in the disposal system is that most of these metals are alloys that do not exist in nature. Consequently the analogues most frequently studied have been of copper, iron, and steel. In the case of copper there are both archaeological and natural analogues, including massive deposits of native copper, while for steel the analogues mainly are limited to archaeological artifacts. Some studies of the corrosion processes of steel have focused on deposits of native iron, which although not exactly the same as the steel produced today, are the only natural analogues known. Studies of the reinforcing rods included in old concrete used in chemically-reducing environments may provide useful information on steel corrosion products in an environment similar to those of the geologic disposal facility. The information obtained from the study of metal analogues is important to assess the durability and longevity of iron and steel and copper, including corrosion modes and rates, and to the properties of the secondary alteration products.

B-4.1.3 Backfill, Buffer and Sealing Materials

B-4.1.3.1 Bentonite

Clay may be used for backfill and buffer materials in the engineered barriers of some geologic disposal concepts; it is also one of the host rock media considered in this study for geologic disposal systems. The clay selected as a buffer material in most mined disposal concepts, and as a sealing material in the deep borehole disposal, is bentonite. The main mineralogical component of bentonite is smectite, which is an expansive clay; it has the capacity to sorb water or organic liquids between structural layers, leading to an increase in volume, which is desirable for creating low permeability barriers. Smectites also have high ion exchange and adsorption capacity and may act as pH and Eh buffers. Other important physical properties include low permeability, which limits groundwater flow, and a high degree of plasticity, which allows the bentonite to flow and seal void space, and moderate thermal conductivity, which allows dissipation of the heat generated within the waste package. The properties of swelling, ion exchange and hydraulic, microbiological, and colloidal isolation make it one of the most important elements the multiple barrier system of some geologic disposal systems.

In a geologic disposal system, the bentonite could be subject to chemical processes that may cause changes in the capacity of the material to perform its functions as a barrier, especially during the thermal pulse, when waste induced heat can cause irreversible dehydration. Other processes that may lead to degradation of the bentonite are chemical precipitation of mineral phases in the pore space or voids in the material due to water-bentonite interactions, and the progressive conversion of smectite into another mineralogical component known as illite. Illite has a lower swelling capacity than smectite and higher permeability. The formation of illite implies a degradation of initial properties of the bentonite.

The study of natural clay systems that have undergone alteration over time as a result of different geological processes may be of great help in understanding a long-term behavior of bentonite. The relevant information that may be obtained from such natural analogues includes characterization of long-term stability, including the analysis of changes in their properties induced by natural thermal processes, for example, that caused by igneous intrusion at the analogue site, the relative importance of diffusion and advection and small scale heterogeneities in transport mechanisms, and molecular diffusion coefficients. Other important information includes the study of redox front movements, including the effects produced by the presence of fractures, and the study of speciation processes affecting radionuclides and other trace elements in the interstitial waters of these materials, including the formation of organic complexes as a result of the presence of organic matter in clay materials.

B-4.1.3.2 Cement and Concrete

Cement and concrete will be used to some extent in the construction of all geologic disposal systems currently being considered. In the case of used nuclear fuel and high-level radioactive waste, the main use will be stabilization of the access tunnels or shafts during the construction phase and perhaps as permanent plugs for the final sealing of tunnels and waste emplacement drifts. Concrete is a key component of the Belgian Supercontainer concept (Wantz et al. 2004), and is also an important part of the engineered barriers for intermediate level waste geologic disposal concepts (Organisation for Economic Co-operation and Development 2003a). Favorable properties of concrete for use as a barrier include its mechanical strength, low permeability, and its role in maintaining a long-term alkaline environment. The interstitial waters in a geologic disposal system that includes significant amounts of concrete will be characterized by very high pH; for this reason they are known as hyperalkaline waters. This property is particularly important to limiting the movement of radionuclides, because the solubility of most radionuclides and therefore their potential for transport are substantially lower in strongly alkaline environments. Another beneficial aspect of these environments is the almost complete inhibition of bacterial activity due to the extreme pH conditions.

The cement and concrete to be evaluated for use in geologic disposal systems are Portland cement based; the main hydration products of Portland cements are calcium-silicate hydrates. These compounds form an amorphous gel that provides a binding force between the cement particles. The calcium-silicate hydrate gels are thermodynamically unstable and spontaneously transform into stable crystalline forms. The rate of this process is too slow to be measured experimentally and cannot be calculated easily, leading to an interest in analogue studies. There are two approaches to the study of concrete and cement analogues. The first is a study of the anthropogenic concrete in archeological buildings or ancient industrial construction. A problem with this type of analogue is that the Portland cement used in current concrete has different properties from those of ancient cements which were based on limestone. However certain older cements also contain calcium-silicate hydrate compounds, which have served to preserve them.

The second approach consists of studying the natural occurrence of minerals analogous to the compounds encountered during hydration of Portland cement. The presence of certain of the minerals in the cement or in hyperalkaline waters, such as interstitial waters found in these materials, is not common in nature. There are, however certain natural systems in which hyperalkaline waters are found, such as those originating as a result of complex water-rock water interaction is in the alteration of ultramafic rocks, of the alteration of thermally metamorphosed limestones and marls. Of greatest interest in examining the analogues are cement longevity and its binding properties, its permeability to water and gas, the speciation and solubility of radionuclides and other elements under high pH conditions, the interaction of high pH fluids with surrounding rocks as an analogue of interstitial waters migrating from the engineered barrier system to the host rock, the nature and viability of microbially induced geochemical processes and their influence on waste form degradation and the subsequent mobilization of radionuclides in the near field, and the nature and stability of colloidal species formed in high pH waters.

B-4.1.4 Geosphere

In geologic disposal systems, a properly selected host rock provides physical isolation from the potential impacts arising from disruptive events and processes, and serves as a barrier against human intrusion. It also provides a stable geochemical and geomechanical environment and limits the quantity of water that potentially could contact the engineered barriers. Furthermore, in the event of engineered barrier failure or degradation, the host rock contributes to retarding the migration of radionuclides released from the engineered barrier system. Geologic formations provide adequate hydraulic characteristics including low permeability and hydraulic gradient, mechanical, geochemical, and seismic stability, and the structural characteristics needed to develop the geologic disposal facility. Host rocks that are suitable for development of geologic disposal facility exhibit long-term stable behavior; in a sense, the host rock itself is the geologic analogue. Understanding what has happened in the past allows for a better understanding of what is expected to happen in the future (Miller et al. 2000).

B-4.2 Analogues Illustrating Disposal System Evolution Processes

The expected behavior and evolution of a deep geological disposal facility for radioactive wastes in granite or clay/shale after closure and sealing ultimately involves the possibility of water entry into the disposal system. This water could have several effects, ranging from causing the bentonite in a buffer to swell and saturate to eventually allowing water to reach the waste package and cause corrosion. Provided the buffer material retains its integrity, the amounts of advective water movement would be expected to be very low, with water movement, and any eventual radionuclide transport, dominated by diffusive flow. The waste package is designed to provide containment of the wastes for a period of time depending on the material characteristics; typically it would be expected to be at least several thousands of years, although with a favorable combination of metal and geochemical conditions it could be much longer (as for the Swedish concept in granite). Eventually, the waste package will fail due to degradation by corrosion.

The radionuclides released from the waste would be dissolved and transported in the groundwater, initially through the buffer materials in the near field for those disposal concepts that include an engineered buffer. Saturated bentonite is practically impermeable to water, and as a result, solute transport occurs principally by diffusion. Chemical processes will affect these materials, possibly leading to radionuclide retardation and retention through mechanisms such as precipitation in the bentonite or sorption, and to changes in the properties of the barriers, especially those caused by the thermal pulse period, including cementation, or transformation of smectite into illite which has poorer sorption and swelling properties.

In the case of fractured media where the fractures are the main pathways for water flow, the radionuclides that reach the geosphere are transported principally by advection and dispersion. In clay/shale formations the main transport mechanism is diffusion. A series of reaction processes takes place resulting in conditioning of the chemistry of the groundwater and possibly leading to radionuclide retardation and retention, including sorption, dissolution or precipitation between the groundwater and the host rock, and diffusion in the matrix. Other processes take place this in the disposal system that can condition and have a major influence on radionuclide release rate, migration, and retention. These include processes associated with the radiation present in the wastes, thermal processes associated with the temperature variations induced by the heat generated by the wastes, processes relating to the properties and mechanical behavior of the engineered and geological barriers, processes associated with multiphase fluid flow, and geochemical processes that determine the evolution of the interstitial waters of the engineered and geological barriers.

In addition to demonstrating long-term robust and stable behavior of the disposal materials and components, an important use of the analogues illustrating disposal system evolution processes is to provide information that is used to verify that the long-term projections made in the safety assessments are reasonable. The use of such information is described in the following sections.

B-4.2.1 Dissolution of the Waste Matrix (Used Fuel or Vitrified Waste from Reprocessing)

The main processes of alteration that may affect borosilicate glasses are devitrification, through which phases of greater solubility than the original glass may be produced, and its dissolution on contact with groundwater following failure of the waste package. Observations of natural volcanic glasses and archaeological glasses have provided qualitative information on the processes and quantitative data on the ranges of dissolution rates. These have indicated long-term stability of these materials under a wide range of conditions.

Chemical alteration of the used light water reactor fuel matrix may arise as a result of simple dissolution or transformation processes such as such as the oxidation of the UO_2 into oxides having different chemical properties and relationships. These processes are related and principally depend on the redox state of the system. The study of the sequence of alteration of the uraninite analogue of the UO_2 matrix in uranium deposits under both reducing conditions such as those at Oklo, Cigar Lake, and Palmottu, and oxidizing conditions such as those that exist at Shinkolobwe, has provided qualitative information on the evolution of used nuclear fuel under different potential geologic disposal system conditions and quantitative data on dissolution rates, which has contributed to the verification of radionuclide release models.

B-4.2.2 Criticality

The accumulation of fissionable isotopes such as ^{235}U and ^{239}Pu in the used nuclear fuel, when in the presence of a neutron moderator such as groundwater, potentially could lead to conditions of criticality in a geologic disposal facility, generating fission products and heat (current U.S. regulations require consideration of this issue and means to prevent its occurrence). In the extreme case of initiation of a rapid chain reaction, damage potentially could be caused to the geologic disposal system. The natural analogue at Oklo, where conditions of criticality were reached about two billion years ago, provides a unique opportunity to study the transport of the radionuclides associated with geologic disposal as well as the stability of uranium minerals. Information from the Oklo analogue has been used to analyze the possibility of criticality in a high-activity waste disposal system, concluding that it is highly improbable.

B-4.2.3 Radiolysis

Radiolysis is the process of chemical alteration of water as a result of exposure to high-energy ionizing radiation. The most important phenomenon of radiolysis in the near field of a geologic disposal facility is radiolysis of the groundwaters, with ionization and electronic excitation and the generation of oxidizing or reducing agents that may have an effect on waste package corrosion, increasing or decreasing corrosion rates respectively, UO_2 dissolution, and radionuclide solubility. The process has been studied in uranium deposits with high radiation fields such as Oklo and Shinkolobwe (Miller et al. 2000), contributing to increased knowledge of its effects on the oxidation and degradation of uraninite and to the development and improvement of models simulating the process.

B-4.2.4 Corrosion

Corrosion involves the degradation of material, principally metals, as a result of reaction with the surrounding environment. Corrosion is the main cause of waste package failure in a geologic disposal system. The corrosion phenomena may occur in a generalized or localized manner, depending on the characteristics of geological media, groundwater, the disposal system, and waste package material itself and the waste package manufacturing and fabrication processes. Corrosion may also give rise to the generation of secondary alteration products and gases. The study of metallic material archaeological analogues such as the Kronan Cannon, the Inchtuthil Nails, the Changsha Tomb, or the Tournai Sarcophagus, provide a way of quantifying the corrosion behavior of materials on a scale of hundreds to thousands of years, and has shown the ability of certain waste package corrosion products to delay radionuclide migration in the near field.

B-4.2.5 Dissolution-Precipitation of Impurities in Bentonite

The dissolution and precipitation of the accessory products present in bentonite such as quartz, feldspars, carbonates, or sulfides, constitutes one of the principal processes determining the geochemical evolution of waters in the bentonite barrier. Certain processes, such as precipitation of pyrite or siderite, favor a reducing environment in the bentonite. Observations on sulfides and ferrous carbonates dispersed in the clay halo at Cigar Lake, which is an analogue for bentonite in a geologic disposal facility, have corroborated the extrapolation of laboratory tests that indicate the importance of these minerals in maintaining a reducing and stabilizing environment.

B-4.2.6 Dissolution-Precipitation Processes in a Variable Temperature Field: Cementation in the Bentonite Barrier

The processes of chemical precipitation of the mineral phases in the pores or voids of the bentonite and their interaction with water due to thermo-hydro-chemical effects are known as cementation. This may affect the rheological properties of the bentonite, increasing its brittleness, reducing its swelling capacity, and potentially diminishing its qualities as a barrier. The study the cementation of siliceous phases in the natural analogues at Kinnekulle has shown that these processes would not appear to be sufficiently important to be relevant to the integrity of the barrier. This confirms laboratory results extrapolated to the timescales of interest for safety assessments.

B-4.2.7 Transformation of Smectite and Illite

The transformation of smectite and illite is a process of mineral replacement in which there is an exchange of the cations for the potassium cation in the interstitial solution. It occurs naturally in geological systems at relatively high temperatures, and might take place during the thermal transient period of the geologic disposal facility. It results in the generation of illite, which is less effective as a barrier material than smectite. Diagenetic processes such as those associated with igneous intrusion as are considered to be analogues for this process. The study of analogue sites such as Kinnekulle has provided data on the intensity or timescales required for the process to occur, confirming that these timescales should be longer than those considered in the safety assessment, and has also led to verification of the kinetic models used to describe the illitization. At Cigar Lake, the clay halo surrounding the uranium deposit is composed principally of illite, so its study makes it possible to analyze the performance of this material as a barrier.

B-4.2.8 Speciation-Solubility: Solubility Limits Calculation

The speciation-solubility properties of radionuclides are specific to each element and depend on the chemical characteristics of groundwater after its interaction and conditioning following interaction with the engineered barriers. The calculation of the solubility limits of the radionuclides in geologic disposal facilities, through the use of geochemical models and thermodynamic databases make it possible to determine the specific concentration of each of the radionuclides during its transport from the waste form to the biosphere. Practically, however, in most safety assessments, solubility limits are typically applied only in the near field, as a measure of conservatism. Natural analogue studies at Oman, Pocos De Caldas, Cigar Lake, Maqarin, El Berrocal, Oklo, and Palmottu have led to confidence in thermodynamic databases and conceptual geochemical models.

B-4.2.9 Cement Degradation: Generation and Evolution of the Hyperalkaline Plume

The degradation of cement includes various states that depend on the characteristics of the medium and that, as a result of interaction between the concrete and the groundwater, condition the development and evolution of the halo of high pH fluids. This hyperalkaline plume affects the near-field materials including the solubility of radionuclides, the corrosion of metals, and alteration of the host rock. The study of archaeological and industrial analogues such as the conduit at Acquarosa, Hadrian’s Wall, and the Tank at Uppsala Castle, or natural analogues such as Oman and Maqarin may provide information on these processes and their effect on radionuclide solubility and the conceptual development of hyperalkaline plume development.

B-4.2.10 Fluid Flow

In geologic disposal systems in fractured granitic rock the flow of groundwater to the geosphere often concentrates in a system of fractures, which act as the preferential flow path; the geometry and configuration of the fractures condition the flow characteristics. These fractures affect the transport and dispersion of the radionuclides, their retardation, through determining the quantity of surface area available for diffusion in the matrix and for sorption, lithological stresses, temperature, and the flow and transport of gas in the rock. Natural analogues studies such as those at Oklo, Cigar Lake, Palmottu, El Berrocal, and Pocos De Caldas may provide information to corroborate the results of field and laboratory studies on aspects such as transit times for water from the geologic disposal facility to the biosphere.

B-4.2.11 Advection and Dispersion

Advection, which is the transport of substances by the motion of the fluid in which it is present, and hydrodynamic dispersion, which is natural mass transport from zones of higher concentration to those of lower concentration of solute due to molecular diffusion and mechanical dispersion, are phenomena that control the transport of radionuclides dissolved in groundwater. Although they are classical processes that have been studied widely in hydrogeology, the study of natural analogues such as those at Oklo, Palmottu, El Berrocal, or Maqarin has provided abundant information on the parameters involved in advective – dispersive transport of solutes. This may provide information to corroborate the results of field and laboratory studies on aspects such as transit times for water from the geologic disposal facility to the biosphere.

B-4.2.12 Molecular Diffusion

Molecular diffusion is the net transport due to movement of substances from zones of higher concentration to lower concentration. It is the main transport mechanism in buffer and sealing materials such as bentonite, and in the geosphere in the case of geologic disposal facilities developed in clay/shale formations. In fractured media such as granite, molecular diffusion of solutes into the rock zone around the fractures occurs and is known as matrix diffusion.

The fundamentals of molecular diffusion in clay/shale media has been studied by means of laboratory and in-situ experiments. The study of analogues for diffusion in bentonite is more limited, although it has been studied in some detail at Cigar Lake. Diffusion in clay formations has been studied in various analogues such as Loch Lomond, Koongarra, and Dunarobba, providing quantitative data for the diffusion of certain species of interest and supporting the conservatism of the coefficients of diffusion obtained in laboratory tests. In fractured formations matrix diffusion acts as a radionuclide sink, causing both a reduction in maximum activities and a delay in radionuclide movement toward the biosphere. It is one of the most widely considered and contrasted processes of natural analogues studies at sites such as Palmottu, Pocos De Caldas, and El Berrocal, allowing progress be made in developing knowledge of the factors important to molecular diffusion, including volume of interconnected porosity, importance of grain boundaries microfractures and the mineralogy the matrix, ranges of values of apparent diffusivity and their uncertainties be studied. The studies also gave insight to the relative importance of diffusion in the matrix compared to other retardation mechanisms.

B-4.2.13 Water Rock Interaction Processes

Water rock interactions include the set of processes that condition the characteristics and evolution of the water that eventually could come in contact with the disposal system barriers. It also includes the evolution of certain mineralogical characteristics of the host rock due to interaction of groundwater, which can influence the geochemical environment and radionuclide retention properties. These are treated jointly in the safety assessments, and are referred to as geochemical processes. They determine not only the performance of the materials in the engineered barriers but also radionuclide transport in the geosphere. The study of these geochemical processes in natural analogues has contributed to the development of conceptual models for the dissolution and precipitation of secondary mineral phases, underlining their importance in the maintenance of hydrogeochemical stability in fractured media such as at Cigar Lake and Pocos de Caldas, and in radionuclide retention studies such as at Palmottu and El Berrocal.

B-4.2.14 Retardation Processes: Sorption and Precipitation / Co-precipitation

The main processes of radionuclide retardation exhibited in interaction with the materials comprising the disposal system, acting to slow down radionuclide movement and reduce the concentrations in solution, are sorption, precipitation and co-precipitation, matrix diffusion, molecular filtration, and ion exclusion. It is very difficult to distinguish one from another and therefore relate a given effect to a single one of these processes. Sorption is the adherence of dissolved chemical species to the electrically charged surfaces of minerals, and generally encompasses sorption and ion exchange. It has been studied in clay formations at Loch Lomond, and in the materials filling fractures in granites which are responsible for retardation and sorption processes in fractured media, at Palmottu and El Berrocal.

Precipitation and co-precipitation are processes of structural immobilization of the solute in a new mineral in the media through which it was being transported. The study of these processes in analogues such as Oklo, El Berrocal, Palmottu, Koongarra, Pocos De Caldas, Tono Mine, and Maqarin has provided evidence supporting the conceptual model of the process.

B-4.2.15 Redox State / Redox Front

The redox state of an aqueous system is defined by the concentrations of all the oxidized and reduced species present. Characterization of the redox state in a natural hydrogeochemical system is a difficult task due to the multiple processes affecting it and the difficulty in determining certain fundamental data such as Eh. Work performed on the characterization of redox processes and natural analogues such as at Palmottu and Oklo, have contributed to the refinement and improvement of the in-situ determination of the redox potential in groundwater, and conceptual models of the redox state evolution.

Redox fronts are created at the boundary of two interacting systems with different oxidation environments. The main causes for the development of a redox front in a geologic disposal system are the oxidizing or reducing agents generated during the radiolysis of water close to the waste package, the introduction of air and oxidizing waters during the phases prior to closure of the geologic disposal facility, and the eventual ingress of oxidizing recharge waters. As the solubility and speciation of many radionuclides are highly conditioned by the redox state of the system, their potential for migration or retardation is affected by crossing a redox front. Their study in natural system such as those in analogues at Pocos De Caldas, Oklo, and Cigar Lake may help to understand the dynamics of redox fronts in different situations, providing information on the processes associated with redox front development, the parameters influencing the evolution of movements of the fronts, and the values of their propagation rates.

B-4.2.16 Colloid Generation and Transport

Colloids are small solid particles (1 µ to 1 nm), with a very high associated specific surface, and dispersed in water in the waste disposal system that may exist in the natural medium or be generated as a result of degradation of the engineered barriers. They may absorb radionuclides and constitute an additional transport mechanism; radionuclides that are sorbed onto the colloid phase may be excluded from possible retardation or retention effects. The study of natural analogues at Cigar Lake, Palmottu, Pocos De Caldas, Koongarra, El Berrocal, and Maqarin has contributed to improved understanding of their generation processes in engineered barriers, the types, concentrations, and stability of colloids in the geosphere, their reactions with radionuclides, and the degree reversibility of these reactions, their mobility, and transport. They have also provided quantitative data on certain processes related to colloidal radionuclide transport.

B-4.2.17 Microbial Processes

In a geologic disposal facility, microbes may be introduced during construction of the facility or may exist in the host rock medium. The viability and possibility of their growth and persistence will depend on the availability of energy sources of nutrients, competition for water with the bentonite, and the radiation field. Microbial activity may contribute to the degradation of the engineered barriers by altering the groundwater chemistry or allowing for select, and perhaps irreversible, radionuclide sorption onto microorganisms. This could potentially enhance corrosion rates, increase the solubility of radionuclides in the near field, reduce their sorption capacity, and favor transport to the biosphere. Studies in analogues at Palmottu, Pocos De Caldas, El Berrocal, Oman, and Maqarin, have focused on the types of microorganisms and their effect of hydrogeochemical and radionuclide migration conditions, the availability of nutrients and energy sources for the microorganisms, and their tolerance to extreme radiation, temperature, and alkalinity conditions, in the near field of the of the disposal system.

Appendix C

Previous Repository Safety Assessments

Appendix C— Previous Repository Safety Assessments

C-1. Introduction

This appendix presents the results of studies that have used quantitative assessments to evaluate the suitability of a deep geologic disposal facility for isolating high-activity waste. The examples include (1) early studies in the United States during the historical development of policies and regulations governing disposal of used nuclear fuel and high-level radioactive waste, (2) generic studies that have been performed recently to investigate the suitability of various geologic media and borehole disposal for disposition of and used nuclear fuel and high-level waste, and (3) studies in other countries, selected because they represent programs that have detailed safety case documents that describe the safety assessments in a variety of different geologic media.

The U.S. examples represent evaluations addressing either the original U.S. Environmental Protection Agency regulations for high-level waste disposal (U.S. Environmental Protection Agency 1985) or those prepared subsequent to the National Academy of Sciences recommendations (National Academy of Sciences 1995). The regulations for compliance in other countries differ. The examples presented herein (1) show that geologic disposal is technically feasible in the variety of media and (2) encompass the likely breadth of possibilities for future U.S. geologic disposal options.

The results of these quantitative investigations and safety assessments show that it is possible to demonstrate that geologic disposal could be done safely in a variety of geologic media or through deep borehole disposal. In addition, the results are generally consistent with the generic postclosure safety assessments presented in Section 4.

C-2. Evaluations of the Expected Performance of Geologic Disposal Systems: Supporting Historical Policy Development

This section presents a brief review of three historical studies that used quantitative assessments to evaluate the suitability of a geologic disposal facility. These studies—the *Environmental Impact Statement on the Management and Disposal of Commercially Generated Radioactive Wastes* (U.S. Department of Energy 1980), the Waste Isolation Systems Panel (National Academy of Sciences 1983), and the Background Information Document for the U.S. Environmental Protection Agency rule 40 CFR Part 191 (U.S. Environmental Protection Agency 1985)—were all early studies in the United States supporting the historical development of policies and regulations governing disposal of used nuclear fuel and high-level radioactive waste. An important aspect of those studies was the identification of criteria that could be important to assessing repository safety.

The Environmental Impact Statement (U.S. Department of Energy 1980) concluded that a nuclear waste repository could be sited, loaded, and sealed with every expectation that long-term radiological impacts would be nonexistent. The Waste Isolation Systems Panel identified two important system performance elements: (1) total containment was provided by the absence of flowing groundwater that could in time come in contact with the waste form; and (2) adequate containment resulted principally from low, solubility-limited release rates of the radionuclide products in the waste form, geologic retardation, and a decrease in potential radiation doses to individuals resulting from the dispersion and dilution processes during transport and on discharge in surface water. Safety assessments conducted by the panel used a conceptual repository designed to contain high-level radioactive waste, transuranic waste, and other special long-lived wastes from used fuel from light water reactors, for salt, granite, basalt, and tuff.

The U.S. Environmental Protection Agency (1985) assembled models for four of the five sites tentatively identified by the U.S. Department of Energy for nomination as potential sites for the first repository. In addition, the agency assembled three models of repositories in granitic formations and presented comparisons of population risks from geologic disposal in repositories in different geologic media.

C-2.1 Environmental Impact Statement on the Management and Disposal of Commercially Generated Radioactive Wastes

As discussed in Section A-3.1, the *Environmental Impact Statement on the Management and Disposal of Commercially Generated Radioactive Wastes* (U.S. Department of Energy 1980) compared a variety of disposal concepts for radioactive wastes and concluded that mined geologic disposal was the preferred option. In considering the long-term performance of mined geologic disposal, the Environmental Impact Statement quantitatively examined four geologic media to illustrate a range of rock properties for a radioactive waste repository: salt deposits (bedded and dome), granite, shale, and basalt. All four rock types were thought to possess properties that would be favorable for waste isolation.

The Environmental Impact Statement considered six factors relevant to geologic disposal.

1. **Depth of Repository below the Land Surface**—It was assumed that a range of from 600 to 1000 m of earth material would exist between the repository and the land surface. This would provide a barrier between the waste and the biosphere and protect the repository from human activities. Dimensions of the host rock were also considered so that the repository will be buffered by rock material laterally and below as well as above it.
2. **Properties of the Host Rock**—The physical, chemical, and thermal properties in the host rock determine its capability to isolate and contain the waste and reduce unwanted interactions between the rock and waste. These possible interactions include radiation effects on the rock and chemical and physicochemical interactions. Important rock characteristics include strength, permeability, thermal conductivity and expansion, and radiation resistance.
3. **Tectonic Stability of the Repository Area and Region**—Proper consideration of this important factor would reduce the likelihood of deformation or disruption of the host rock and thus increase the probability of repository integrity.
4. **Hydrologic Regime (i.e., surface water and groundwater considerations)**—This was considered important because the existence of connected water channels could provide potential pathways for waste transport away from the repository.
5. **Resource Potential of the Repository Site and Area**—A low resource potential was desirable to avoid loss of an economic resource by the existence of a repository and to reduce the likelihood of future exploration activities for resource recovery.
6. **The Multi-Barrier Safety Feature**—This combined the redundant isolation features provided by the rock properties, the geologic setting, and engineered barriers to give overall added confidence that the waste would remain isolated.

In general, the most important factors were thought to be the hydrologic regime, the tectonic regime, the multi-barrier concept, and thermal, physical, and geochemical properties of the host rock.

The Environmental Impact Statement considered the following criteria for repository site selection would be used to ensure that the natural barrier functioned as planned:

- The repository site should be located in a geologic environment with geometry adequate for repository placement
- The repository site shall have geologic characteristics compatible with waste isolation. The repository site shall have subsurface hydrologic and geochemical characteristics compatible with waste isolation.
- The repository site shall be located so that the surficial hydrologic system, both during anticipated climatic cycles and during extreme natural phenomena, shall not cause unacceptable adverse impact on repository performance

- The repository site shall be located in a geologic setting that is known to have been stable or free from major disturbances such as faulting, deformation, and volcanic activity for long time periods
- The repository site shall be located in an area that does not contain desirable or needed mineral resources, or to the extent presently determinable, resources that may become valuable in the future

The repository conceptual design used borehole emplacement of the waste packages. These boreholes would have been backfilled, and the backfill materials were to be designed to fulfill one or more of several functions.

- Sorbing the limited amount of water that may be present in a repository rock, e.g., from brine inclusion migration in salt
- Impeding the movement of intruding groundwater to and from the waste package
- Selectively sorbing radioisotopes from groundwater in the event of canister breach
- Modifying groundwater chemistry and composition (e.g., pH, Eh) to reduce corrosion rates or minimize waste form leaching
- Providing mechanical relief by accommodating stresses on the waste package induced by rock movement
- Serving as a heat transfer medium

The Environmental Impact Statement concluded that a nuclear waste repository could be sited, loaded, and sealed with every expectation that long-term radiological impacts would be nonexistent. The Environmental Impact Statement did, however, consider a few highly improbable events that could be postulated to take place that might result in radioactive releases reaching the biosphere. Three kinds events leading to release of some of the repository contents were considered: (1) direct release of the contents to the atmosphere, following, for example, volcanic activity, impact of a large meteorite or large nuclear weapon, or, on a much longer timescale, denuding of the Earth to the depth of the repository by erosion or glaciation; (2) release via water, that might enter a repository as a result of flooding or seepage following the breach of the overlying rock by such mechanisms as fracturing by faulting, nearby impact of a meteorite or nuclear weapon, thermal stresses caused by decay heat from the radioactive waste, mechanical stresses resulting from adjustment of the repository rock following excavations, or failure of shaft or boreholes seals; and (3) release from inadvertent intrusions, which might include exploratory drilling, solution mining for salt or phosphates, or cavern construction for storage of oil.

While the radiation eventually released by such an accident scenario appears as quite large values, these types of scenarios are typically characterized by a low probability of occurrence.

C-2.2 Waste Isolation Systems Panel

A study of the isolation systems for geologic disposal of radioactive wastes was conducted by a panel of the Board on Radioactive Waste Management of the National Research Council (National Academy of Sciences 1983). That panel examined alternative technologies available for the isolation of radioactive waste in mined geologic repositories, evaluated the need for and possible performance benefits from these technologies as potential elements of the isolation systems, and identified appropriate criteria for choosing among them to achieve satisfactory overall performance of a geologic repository. The panel's report summarized a conceptual design for several different repository types and examined construction and operational aspects of the underground facility. The report typically summarizes work that was going on in the late 1970s and early 1980s by the U.S. Department of Energy and its contractors.

The panel identified two key system performance elements: (1) total containment was provided by the long-term absence of flowing groundwater that could in time come in contact with the waste form; and (2) low velocity flowing groundwater was present and adequate containment resulted principally from low, solubility-limited release rate of the radionuclide products in the waste form, geologic retardation, and a decrease in potential radiation doses to individuals resulting from the dispersion and dilution processes during transport and on discharge in surface water.

The panel concluded that, with respect to the current options considered, the technology for geologic waste disposal had advanced to the state of a preliminary technical plan, suitable for testing and for further technical studies and pilot facility confirmation. Information available or soon to be available was thought to be sufficient for the selection of one or more candidate sites for in-situ testing. Following site selection, detailed exploration underground and testing of candidate sites were considered likely to provide sufficient technical information to proceed with detailed design and construction of a repository. Technology that was not yet fully available could provide additional options in the future that would rely on containment within a highly insoluble waste form contained within a longer lived waste package.

The panel concluded that the most meaningful and useful performance criterion was the annual or lifetime radiation dose to an individual exposed at some future time to radionuclides released to the environment from a geologic repository. The criterion used by the panel was an annual radiation dose of 10^{-4} sievert, from a repository containing 100,000 MTHM, to an individual averaged over a lifetime, calculated for all future times. The panel report included a comprehensive evaluation on precedents for the use of individual dose criteria.

Performance analyses, or safety assessments, conducted by the panel used a conceptual repository design for salt, granite, basalt, and tuff, designed to contain high-level radioactive waste, transuranic waste, and other special long-lived wastes from used fuel from light water reactors. The reference case was based on waste from reprocessing fuel 160 days after discharge from the reactor and storing the resulting waste for 10 years prior to emplacement in repository. They also considered other alternatives for the fuel cycle operations.

- Emplace unreprocessed fuel in the repository after first storing it for ten years
- Store the discharged fuel for realistically longer periods prior for subsequent reprocessing
- Store the reprocessing wastes for longer periods prior to emplacement in the repository
- Reprocess fuel from light-water and fast breeder reactors that have been loaded with uranium and recycled plutonium
- Reprocess uranium – thorium fuel

The report found that for repositories in hard rock, the major question of long-term performance is related to the prediction of the rock mass permeability or, specifically, permeability of fracture systems. The permeability of the major hard rock types under consideration is low compared with that of the discontinuities; the report concluded fracture permeability is 6 to 10 orders of magnitude greater than the matrix permeability. The thermal mechanical and thermal chemical response of the fracture system were be deemed to be of primary importance. The report concluded that the natural factors important to hydrologic transport of radionuclides are mainly low solubility of the waste form such that the rate of release of these radionuclides to the environment is low; solubility of each radionuclide is affected by its geochemical properties in the geochemistry of associated waters. Sorption is constrained by the geochemistry of each element, the minerals such as clays and zeolites in the rock, and the properties of associated pore waters, including pH, Eh, and dissolved species. The lack of moving groundwater in the repository host rock, or a sufficiently long time of water travel from the waste to the environment is also desirable.

The report presents calculations of individual radiation dose as a function of water travel time in salt, granite, and tuff for wastes from reprocessing, with sensitivity studies looking at solubility limited solution and dispersion. The report also calculated individual radiation doses as a function of the groundwater travel time for basalts; processes for used nuclear fuel include congruent dissolution, and solubility limited dissolution. The report also includes a section on natural analogues relevant to geologic disposal.

The report presents a detailed summary of the generic characteristics of a range of candidate host rock types: bedded salt, salt domes, granite, basalt, rhyolite (tuff), unsaturated alluvium over a regional aquifer, and granite/metamorphic rocks under a regional sedimentary aquifer. Because site-specific data for granites were not available at the time of the study, generic average properties were adopted for the calculation.

C-2.3 U.S. Environmental Protection Agency

In the Background Information Document for the final rule 40 CFR Part 191 (U.S. Environmental Protection Agency 1985), the U.S. Environmental Protection Agency presented risk assessments for disposal of high-level waste in geologic repositories.

These risk assessments were done to support development of the regulatory standard for high-level waste and transuranic waste repositories. The U.S. Environmental Protection Agency simulated the behavior of repositories with geologic media characteristics comparable to those tentatively under consideration by the U.S. Department of Energy under the Nuclear Waste Policy Act as potential sites for the first repository.

The U.S. Environmental Protection Agency assembled models for four of the five sites tentatively identified by the U.S. Department of Energy, for nomination as the potential sites for the first repository. These four sites are (1) bedded salt in the Palo Duro basin of Texas, (2) the bedded salt deposits in the Paradox formation of Utah, (3) the basalt flows on the Hanford reservation in Washington, and (4) the unsaturated volcanic tuff at Yucca Mountain in Nevada. In addition, the agency assembled three models of repositories in granitic formations. Two of these attempted to roughly represent geological and hydrological conditions that might be encountered in certain regions of the country: one was the model for the north-central portion of United States (granite I), and the other was a model for the New England area (granite II). For comparison U.S. Environmental Protection Agency also used an idealized model of a granite repository that was considered in its proposed rule 40 CFR Part 191 (granite III). This model is typical of the objectives that might be set in looking for a very protective site in granite and may represent conditions that could be difficult to find; however the agency believed that the third model provided a useful perspective because it indicated a range of performance attributable to repositories in granitic media.

The conceptual framework of the potential repository is located between two aquifers, called respectively the upper aquifer and the lower aquifer to simulate conditions present at real sites. The aquifers do not generally represent single hydrostratigraphic units, but rather represent synthetic aquifers with properties defined to approximate the combined properties of a number of transmissive units above and below the repository horizon. The absence of an aquifer either below or above the repository could be simulated, as well as an upward or downward gradient between the lower and upper aquifer. The parameters needed to describe the geometry hydrologic conditions assume for each of the seven repository models are described in the report.

The general way in which risk was modeled was that the upper aquifer was considered to be the pathway for groundwater release of any radionuclides from the repository. The computer codes used in the risk assessments are described in the document. The primary code calculated the consequences and the probability of each of the release scenarios considered, and it combined those estimates into the total expected number of health effects caused over a period of time. It did not calculate individual exposures nor could it adequately estimate integrated releases of radioactivity for more than about 10,000 years.

The U.S. Environmental Protection Agency noted that while the Department could modify its conceptual designs for repositories, an examination of their risk analysis models indicated that they were not highly sensitive to the engineering assumptions. Therefore the U.S. Environmental Protection Agency adopted a single set of design assumptions to be applied for all potential types of sites. The repository was assumed to contain waste corresponding to 100,000 metric tons of nuclear fuel. The repository area encompassed 4,000 m × 2,000 m with a mined out volume of 10^7 m^3. There were 35,000 canisters of waste placed in the repository. These repositories consisted of underground mines or excavations with working levels between 300 and 1,000 m below the ground surface. The radioactive wastes consisted of used nuclear fuel or solidified reprocessing waste in a relatively durable form such a borosilicate glass. The waste would be packaged in canisters that would be placed in holes in the walls or floors of the mined rooms in the repository. After emplacement of the waste, the repository would be backfilled to enhance mechanical stability and retard movement of fluids.

The waste package was assumed to consist of two main components: the waste form and the waste canister. For the conceptual analysis, the Agency adopted a simple model for the waste package with a stable homogeneous waste form and a single canister. The model considered normal groundwater flow, fault movement, breccia pipe formation, inadvertent intrusion by exploratory drilling, and volcanism.

The U.S. Environmental Protection Agency examined the population risks from disposal in geologic repositories and how they would vary with different assumptions about the waste package lifetime and waste form release rates. Waste package lifetimes of 0 and 1000 years were considered for population risks. The waste form release rate varied from one part in 1000 per year to one part in 1 million per year. For granite repository models, variations over the range of waste package lifetimes considered had relatively little effect on the population risks. However, in the bedded salt models, when a very short canister lifetime was coupled with a rapid release rate, it allowed some of the short-lived fission products to be brought to the land surface by inadvertent intrusion.

C-3. Evaluations of the Expected Performance of Geologic Disposal Systems: Supporting Safety Assessments

This section presents a selection of the evaluations and supporting safety assessments conducted by the United States as well as the international community. The selections are organized by the four geologic disposal options: salt (Section C-3.1), clay (Section C-3.2), granite (Section C-3.3), and deep borehole (Section C-3.4). A general description of these four disposal options including their potential advantages and availability is presented in Section 2.2.2.2. The subsections below describe aspects of the initial state and expected evolution in the context of the evaluation or safety assessment conducted for a specific disposal program. These site-specific descriptions inform the generic initial state and expected evolution discussions presented in Section 4.2.3.2 for the four generic disposal options.

C-3.1 Salt

C-3.1.1 Review of Salt Repository Science (United States)

In 1957, the National Research Council of the National Academy of Sciences released a well-known study concluding that salt showed significant promise as a potential medium for high-level radioactive waste disposal (National Academy of Sciences 1957). Research conducted in the fifty plus years since that study has resulted in a considerable body of knowledge regarding the waste isolation capabilities of salt.

In 2011, Sandia National Laboratories published a high-level review of the state of salt repository science and, building on that information, proposed a framework for evaluating a generic salt repository (Hansen and Leigh 2011). In conducting this study, Sandia was able to apply lessons learned from its previous experience on the Waste Isolation Pilot Plant (WIPP) (Section C-3.1.2) and other geologic repositories as well as related salt experience with the Strategic Petroleum Reserve.

In the early stages of repository development decision makers have questions about (1) what factors are important to repository performance and (2) what data or knowledge gaps exist that would benefit from further research. As a result, the proposed evaluation framework is an iterative process featuring safety assessment analyses and a directed science and testing program (Hansen and Leigh 2011, Section 3). This framework is virtually the same as that described in Section 2.3.1. The process begins with defining the performance goals, characterizing the system, and identifying scenarios for analysis in preparation for the safety assessment analyses. Once the safety assessment analyses are done, the results are used to inform a directed science and testing program. Information from this program is in turn used to inform the next iteration of safety assessment analyses.

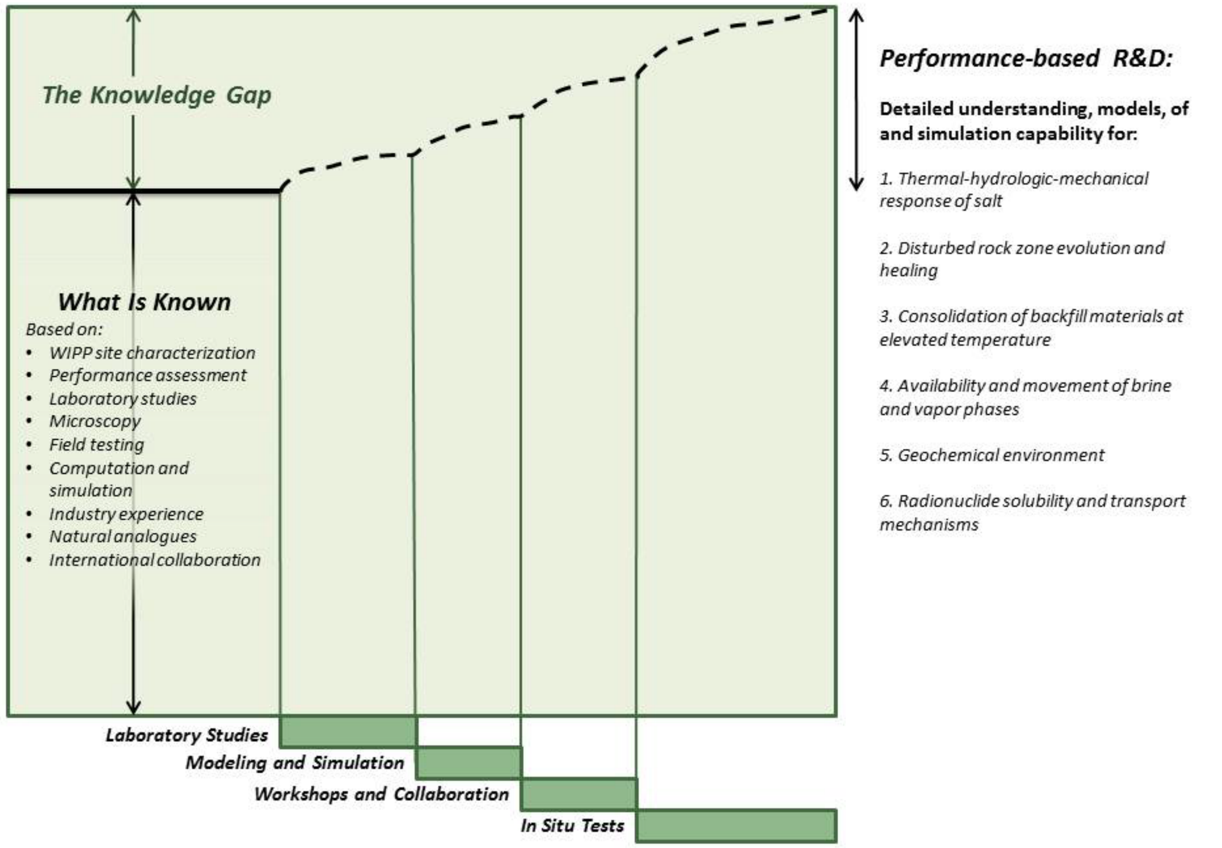
The study discusses two general categories for possible scenario development: an isothermal “cool” salt repository and a thermal “hot” salt repository (Hansen and Leigh 2011, Section 3.2). For the former, the experiences gained from the WIPP can be used to advantage, both in terms of identifying relevant features, events, and processes (FEPs) and scenarios. For the latter, the current knowledge base relies heavily on advancements from programs in other countries, particularly that of Germany. Given the current information, an undisturbed salt repository with hot waste may be even more robust than that certified for non-heat-generating waste.

The study confirms that uncertainty remains in the most important FEPs, and it presents a preliminary gap analysis of what is known versus what is unknown (Hansen and Leigh 2011, Section 4). Figure C-1 is a stylized illustration of the gap analysis, with the following six research areas identified on the right:

1. Thermal-hydrologic-mechanical response of salt
2. Disturbed rock zone (DRZ) evolution and healing
3. Consolidation of backfill materials at elevated temperature
4. Availability and movement of brine and vapor phases
5. Geochemical environment
6. Radionuclide solubility and transport mechanisms

Besides discussing the data gaps in these six areas, the study incorporates the findings of the U.S./German Workshop on Salt Repository Research, Design, and Operation held in May 2010 (Sandia National Laboratories 2010). While the workshop results are broader in perspective than the six research areas, there is significant overlap between the two.

An identified gap does not necessarily mean that it should be addressed in the science and testing program. The gap may be in an area that is not important to repository performance, and thus does not warrant further investigation. Coupling a gap analysis with safety assessment analyses allows for a directed, performance-based approach to the research and development efforts.



Source: Hansen and Leigh 2011, Figure 10.

Figure C-1. Gap Analysis for a High-level Waste Repository in Salt

The following summarizes some of the important conclusions from the study (Hansen and Leigh 2011, Section 5.1):

- The United States has large amounts of land with salt formations of sufficient depth, thickness, and lateral extent for hosting a high-activity waste repository. The positive qualities of salt and the abundance of potentially suitable salt formations in the United States are discussed further in Section 2.2.2.2.1 and Figure 2-4.
- Salt has been recognized as a suitable medium for high-activity waste disposal by national repository programs in the U.S. and other countries. Advancements in engineering, scientific, and operational concepts have increased confidence that a repository in salt could be safely constructed, operated, and sealed. The national programs in other countries responsible for these advancements have much to offer the United States in the investigation of salt as a possible high-activity waste disposal medium.
- Radionuclides are not expected to migrate from the disposal horizon based on thermal, hydrologic and geochemical considerations. Intact salt is impermeable and the fractures are self-healing. The majority of radionuclides in the current waste inventory will be thermodynamically stable as solids and as such will resist migration. Much of the inventory will decay before a human intrusion would likely occur.

- Three-dimensional multi-physics modeling is poised to exploit massively parallel computational hardware, thereby advancing capabilities in safety assessment modeling and field test development. The combination of new computational approaches and hardware will allow scientists to simulate coupled behavior (e.g., thermal-hydrologic-mechanical processes) in salt over long time spans.
- Laboratory testing of intact and granular salt will provide data to enhance phenomenological understanding and parameters for use in thermal-hydrologic-mechanical models.
- An appropriate field test will be needed at some point to prove the principles of the disposal concept and to validate the coupled process models.
- Experience with seal systems used in the WIPP repository will provide significant support to design, construction, testing, and safety assessment of a high-activity waste salt repository.
- A high-activity waste repository in salt is expected to exhibit excellent performance.

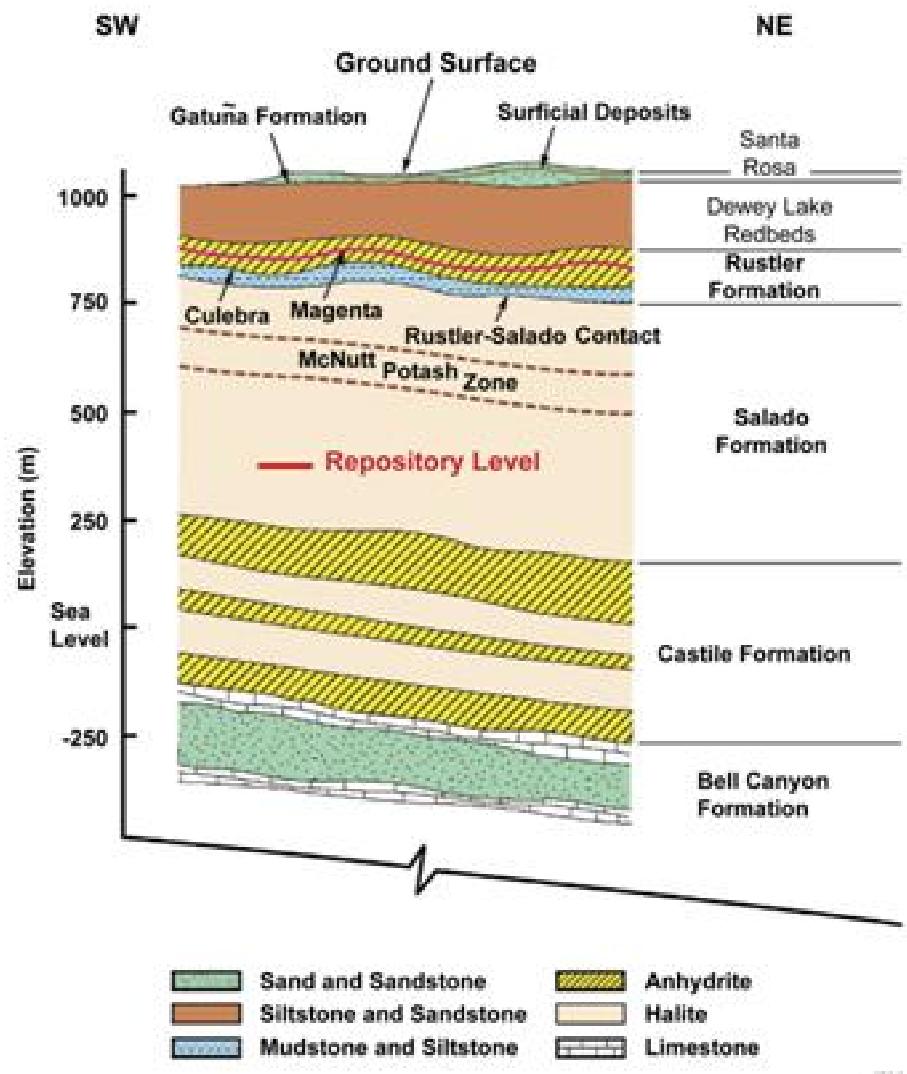
Given the absence of a specific site, current investigations into disposing of heat-generating nuclear waste in a salt repository are generic in nature. Should a specific site be selected, the expectations and assumptions regarding the evolution and performance of such a repository can be examined in greater detail with site-specific safety assessments and a performance-based, directed science and testing program.

C-3.1.2 Waste Isolation Pilot Plant Recertification (United States)

Located 26 miles southeast of Carlsbad, New Mexico, the WIPP is a deep geologic repository in bedded salt operated by the U.S. Department of Energy for the disposal of defense-generated transuranic wastes (U.S. Department of Energy 2009, Executive Summary). Transuranic waste generally consists of items contaminated with radioactive elements, mostly plutonium, such as clothing, tools, rags, residues, debris, and soil. Disposal operations began at WIPP in 1999.

For the undisturbed performance scenario, there must be a reasonable expectation that the annual committed effective dose equivalent will not exceed 15 mrem (150 μSv) for 10,000 yr (40 CFR Part 191.15). For groundwater protection, the Safe Drinking Water Act regulations of 40 CFR Part 141 apply (as they existed on January 19, 1994).

The host rock for the WIPP repository is a 600-m-thick bed of halite known as the Salado Formation, as shown in Figure C-2. Beginning about 250-m below the surface, this salt bed was formed approximately 250 million years ago by the evaporation cycles of the ancient Permian Sea. Below the Salado Formation is the Castile Formation, which consists of a combination of anhydrite and halite layers, and above is the Rustler Formation, which consists mainly of interbedded layers of mudstone/halite, anhydrite, and dolomite.



Source: U.S. Department of Energy 2009, Figure HYDRO-2.

Figure C-2. General Stratigraphic Column of Geologic Units at WIPP

Because the salt is virtually impermeable, water within the Salado Formation is limited to the relatively small amounts trapped millions of years ago. Much work has been done to understand the water movement above the salt bed in the Rustler Formation, particularly in the Culebra (U.S. Department of Energy 2009, Section PA-2.1.4.6). The Culebra, a dolomite, is the most transmissive unit above the Salado Formation. While brine flow in the Culebra has some regional variation, it is generally southward in the area overlying the repository. An ongoing monitoring program continues to add to the knowledge base. The salt bed also has the advantage of long-term geologic stability. The waste will lose most of its harmful radioactivity within a few hundred thousand years, which is short compared to how long the salt has already been stable.

The WIPP repository is located about 655 m (2,150 ft) below the surface. Transuranic wastes are placed in disposal areas and surrounded by backfill. Characterization of the inventory includes information about the types of materials in the waste (e.g., wood, metal, soil), the materials used to package the waste (e.g., steel drums and plastic liners), the emplacement materials (e.g., cellulose, plastic, and rubber), radionuclides in the waste, and chemicals in the waste expected to have some degree of influence on repository performance (U.S. Department of Energy 2009, Section 24.2).

The backfill consists of bags of magnesium oxide (MgO) (U.S. Department of Energy 2009, Section 44.6). The MgO is designed to reduce actinide solubilities by consuming the carbon dioxide (CO_2) that might be produced by microbial activity, an uncertain process. By ensuring low CO_2 fugacity and controlling pH, the MgO backfill reduces uncertainty in repository chemical conditions. Other design features of the repository include shaft seals, panel closures, and borehole plugs.

Through field investigation, laboratory experiments, and safety assessments, the U.S. Department of Energy has developed an understanding of the disposal system and the possible future interactions of the repository, the waste, and the surrounding natural system. Since 1996, five different safety assessments have been documented and submitted to the U.S. Environmental Protection Agency (U.S. Department of Energy 2009, Appendix PA, Section PA-2.1). The latest was submitted in 2009.

For the 2009 safety assessment, 245 FEPs were evaluated: 70 natural FEPs, 61 human-initiated FEPs, and 114 waste and repository FEPs (U.S. Department of Energy 2009, Section 32.6.1). From the FEPs that remained after the screening process, scenarios for consequence analyses were developed representing both undisturbed and disturbed repository performance (U.S. Department of Energy 2009, Section PA-2.3.2). Studies showed that no potentially disruptive natural events or processes are sufficiently likely to occur to warrant inclusion in the undisturbed or the disturbed performance scenarios. As a result, the undisturbed performance scenario includes the retained FEPs that are naturally occurring and non-disruptive. The human intrusion activities that distinguish among the disturbed performance scenarios are mining, deep drilling, or a combination of the two within the controlled area or the land withdrawal boundary.

Mining currently occurs outside the controlled area in the Delaware Basin and may continue to occur in the near future. As a result, mining outside the controlled area and its potential effect on the groundwater flow in the Culebra are considered in both the undisturbed and disturbed performance scenarios (U.S. Department of Energy 2009, Section PA-2.1.4.6). The other mechanism that could potentially affect the groundwater flow in the Culebra during the regulatory period is climate change, which is also taken into account.

The 2009 safety assessment demonstrated that the expected performance of the undisturbed WIPP repository results in effective isolation of the radioactive waste for 10,000 years. The dominant factors affecting the undisturbed system behavior are fluid flow, rock deformation surrounding the excavation, and waste degradation (U.S. Department of Energy 2009, Section PA-2.1.3). These processes are coupled such that the extent to which each occurs is affected by the others.

In the absence of a repository, fluid flow through the host rock from the far field does not occur because of the extremely low permeability of the evaporites in the Salado Formation. This lack of fluid flow is one of the primary advantages of using salt as a disposal medium. Once excavation began, rock deformation changed the natural hydraulic gradient and properties surrounding the repository. Fracturing caused by stress relief created a damaged zone of increased permeability and porosity possibly extending a few meters into the rock. The hydraulic potential within the repository became less than that of the far field, as evidenced by slow brine seeps observed in the repository. The brine originates from the small amount of interstitial water contained in the Salado.

Over time, however, salt creep will tend to heal fractures and reduce the damaged zone permeability. The ability of salt to creep is another one of the fundamental advantages of using salt as a disposal medium. The damaged zone immediately surrounding shaft seals and panel closures will experience particularly rapid healing because these structures provide rigid resistance to salt creep. In addition, salt creep will cause the crushed salt component of the shaft seals to compact such that the properties approximate that of the host rock salt within 200 years. Prior to that time, fluid flow through the shafts is prevented by the concrete, clay, and asphalt components of the shaft seal system.

Nonetheless, the scenario analyses in 2009 safety assessment conservatively assume that the damaged zone does not heal and that pathways created for fluid flow between the repository and the overlying and underlying anhydrite marker beds continue to exist indefinitely. Accordingly, brine in the repository is expected under most conditions and that brine may contain actinides in the form of dissolved and colloidal species.

Gas generation caused by waste degradation processes also impacts the expected performance. Anoxic corrosion produces hydrogen as water is broken down to oxidize steels and other iron-based alloys. Note that, to the extent that all of the brine making contact with the susceptible metals is consumed, the corrosion process will be self-limiting. Anaerobic microbial activity is expected to produce carbon dioxide, hydrogen sulfide, and methane. It is assumed that microbial reactions do not cause a change in the net amount of available water.

To account for the initial air trapped at the time of closure as well as gas generated from waste degradation, fluid flow is treated using a two-phase approach. In response to gas generation, creep closure, and brine inflow, the pressure of the gas phase will increase, thereby inhibiting brine inflow and salt creep. While pressures may approach lithostatic, sustained pressures above lithostatic will not occur because the more brittle anhydrite layers will tend to fracture and allow gas to move away from the repository. Even if such fracturing occurs, gas migration away from the repository is not expected to contribute to actinide release. Brine migration may result in actinide transport through the interbeds to the accessible environment boundary; however, the amount of actinides involved is insignificant and has no effect on the compliance determination. No vertical migration through the Salado is expected to occur.

Under the disturbed conditions, the expected performance of the repository is initially the same as for the undisturbed conditions (U.S. Department of Energy 2009, Section PA-2.1.4). The processes described above are active regardless of whether an intrusion of some kind has occurred. In the advent of an intrusion, additional processes will occur that are not present in an undisturbed repository.

While radionuclide releases to the accessible environment are possible given a disturbed repository, results from the 2009 safety assessment confirm that the releases will remain below the release limits (U.S. Department of Energy 2009, Sections PA-7.0 to PA-9.0). The major contributors are expected to be direct releases to the ground surface because of drilling. These direct releases, which occur at the time of drilling, consist of cuttings, cavings, spallings, and direct brine releases. Other release mechanisms were evaluated, but were not found to be major contributors.

The results from the 2009 safety assessment as well as the previous assessments and continuing scientific studies have shown that WIPP is operating and performing as expected. While WIPP disposes of transuranic waste, the results can still provide useful insights regarding the potential behavior of a salt repository for the disposal of used nuclear fuel and high-level radioactive waste. The thermal period caused by the heat-generating waste would be short compared to the likely regulatory timeframes. The associated processes could cause even quicker salt consolidation and dry out in the surrounding media. The total water budget and the repository end state after complete consolidation would likely be similar to that expected at WIPP. Furthermore, the natural barrier system would be expected to maintain its integrity regardless of the existence of a thermal pulse.

C-3.1.3 Gorleben Preliminary Safety Assessment (Germany)

Germany has been investigating salt as a possible host rock for geologic disposal of high-level waste since the 1960s (Weber et al. 2011). In 1977, the German federal government directed that the salt dome at Gorleben be investigated as a possible site for a geologic repository. However, the government placed a moratorium on the Gorleben site investigation in 2000 and redirected efforts towards the clarification of non-site-specific questions.

As a result, the ISIBEL project was launched to examine situations typical for a salt dome in northern Germany, with the emphasis being on the demonstration of safe containment. ISIBEL is an acronym formed from the German name for the project “review and evaluation of the instruments for assessing safety of HLW repositories.” A preliminary safety assessment based on the ISIBEL concept was begun in 2010 and is expected to be completed by the end of 2012. The assessment addresses a reference period of 1,000,000 yr. Once it is completed, an international peer review will be conducted.

Because of the wealth of information available, Gorleben was used as a reference site for the purposes of building a geologic model for the preliminary safety assessment. The salt dome at Gorleben was formed from Permian-age salt by deformation primarily in the Cretaceous and Tertiary periods. Before development of a domal structure, the primary thickness of the salt layers was approximately 1080 to 1330 m. Internally, the salt dome is marked by significant folding. As a result of the uplift, the once-competent strata of the Main Anhydrite were broken into isolated blocks that float on top of the impermeable rock salt.

While various brines and gases have been found within the salt dome, the potential waste emplacement areas at the domal core are almost completely free of these brine and gas inclusions. The fluids, which are at least 100 million years old or more (i.e., older than the Late Cretaceous), appear to have been fixed in place in specific layers during the time of uplift.

In terms of tectonics, the reference site has been stable since the later Tertiary. During the reference period, only slight subsidence movements are expected. The total uplift caused by diapirism is expected to be about 10 m.

The rocks above and adjacent to the salt dome were developed in different facies from the Triassic epoch to the Tertiary. Their structure was impacted by the salt uplift. Dating from the Quaternary period, the upper layers are sandy and clayey. A thick glacial channel cuts through older formations and in some areas lies directly above the salt. Depending on changing conditions during the reference period, leaching of the salt may occur, but the expected average rates are in the range of only several hundredths of a millimeter per year over extended geological periods (BMWi 2008, Annex 10).

Hydrogeologic investigations above and adjacent to the salt dome have revealed the presence of a system of aquifers and aquitards (Weber et al. 2011). In general, aquitard clay layers hydraulically separate an upper aquifer from a lower aquifer. The exception is in the south-east and north-west where some gaps in the separating layers allow contact between the two aquifers.

The high-level radioactive waste being considered for deep geologic disposal comes from reprocessing of spent fuel elements. This waste includes (1) vitrified high-level radioactive fission products and feed sludge, (2) vitrified medium-level radioactive decontamination and rinsing waters, and (3) compacted medium-level radioactive fuel element cartridges, structural parts, and technological waste.

The Germans are pursuing a strategy of permanent disposal of the waste in salt; retrievability is not part of the plan (EPRI 2010, Section 7.8). Acknowledging the importance of the geologic barrier, the safety concept dictates that the design, layout, and operations be planned such that the integrity of the geologic barrier can be demonstrated (Weber et al. 2011). For example, the design must ensure sufficient depth of the emplacement cavities; it must also maintain an appropriate distance from potential fault zones or strata boundaries. In addition, the maximum surface temperature of a waste package in contact with the surrounding salt must not exceed 200°C. This boundary condition is based on the thermal stability of different salt types.

The reference design places the repository horizon at a depth of 870 m. Two alternative mine layouts were provided for use in the safety assessment. One version uses a combination of horizontal drifts at the 870-m horizon and vertical boreholes drilled 300-m deep from the horizon depending on the type of disposal container. The other version uses vertical boreholes regardless of the type of container.

The safety concept does not rely on the waste containers to act as an engineered barrier during the postclosure period. In addition, there are no plans for additional measures to retain radionuclides in the event that brines reach the waste.

Two engineered barriers—seals and backfill—are included in the safety concept. Shaft and drift seals will be used to seal the shafts and access drifts to the emplacement areas. As part of the preparations to close the repository, crushed salt will be used to backfill all the void volume throughout the mine workings including all boreholes, drifts, and galleries.

The approach for the current safety assessment differs from that used in the past. Previous long-term safety assessments in rock salt typically addressed conservative release scenarios with no consideration of how likely or unlikely the developments within the repository system are to occur. The main goal of this approach was the elaboration of conservative release scenarios, not the systematic demonstration of safe confinement.

The difficulty with the past approach is that it does not fully incorporate the advantages of salt in terms of its ability to perform as a geologic barrier. In contrast, the current safety assessment approach emphasizes the demonstration of long-term safe confinement of the radioactive waste based on the demonstration of the integrity of the geologic and engineered barriers.

The domal rock salt has multiple characteristics that enhance its expected performance as a geologic barrier. As mentioned previously, with the exception of some brine inclusions located away from the potential emplacement areas, the domal rock salt is dry. In addition, it is virtually impermeable and self-healing. As a result, neither pressure-driven advective flow nor concentration gradient-driven diffusive flow can occur.

Rock salt can become more permeable if applied stress exceeds the dilatancy limit causing microcracks to form within the salt. However, salt creep will tend to heal fractures and reduce permeability. According to laboratory analyses of the petrophysical properties of rock salt, stress states below the dilatancy limit do not cause any damage, even over long timeframes. Once a specific site is chosen, geomechanical predictive models can be used to examine further the physical processes that influence the stress state of the host rock during the reference period.

Over time, the crushed salt serving as backfill will exhibit barrier performance capabilities similar to that of the rock salt. The convergence of the surrounding rock salt will compact the crushed salt resulting in decreased porosity and permeability. Until this time, the shaft and drift seals will preclude the possibility of brine intrusion or radionuclide transport through the shafts and drifts.

Because of the positive barrier qualities of the selected reference site as well as the generic design and layout, possible release scenarios typically involve FEPs that have a very low probability of occurrence. Safety requirements published by the German Federal Environmental Ministry provide guidance on the probability limits used to determine the necessity of considering a particular scenario. As part of the ongoing safety assessment effort, various investigations are being conducted regarding expected performance given some type of barrier failure.

One of the major areas of study with respect to the engineered barriers is the potential failure of seal construction. The model results indicate that the failure of a single seal does not cause any radionuclide release during the reference period of 1,000,000 yr. This result is true whether the failure is in a shaft seal of some other seal between the infrastructural part and the access drifts. A shaft seal failure has a greater impact on model results than a failure of a seal in the interior of the repository mine workings. The only scenario resulting in water flow to the emplacement areas and radionuclide transport to the geosphere involved the failure of both types of seals at once.

In an effort to test the methods and tools, a special what-if scenario was defined with respect to the potential for interaction between brine inclusions and emplacement boreholes. It assumed the presence of six brine inclusions in the host rock. Each inclusion had a volume of 100 m^3 and was in contact with an individual borehole. The model results always show some amount of radionuclide release as liquid entering the emplacement areas becomes contaminated and is pressed out of the repository because of convergence. The amount of radionuclides released during the reference period is very low if the sealing constructions are intact. If there is inflow from brine inclusions and all the sealing constructions fail (worst case), the model results show significant release and exposure. However, the probability of occurrence for this combination of FEPs is far below the limit of consideration established by the German Federal Environmental Ministry.

In summary, the safety concept being pursued by the Germans emphasizes the systematic demonstration of long-term safe confinement of high-level radioactive waste by demonstrating the long-term effectiveness and integrity of the geological and engineered barriers. In addition, potential release scenarios involving some impairment of barrier integrity (e.g., seal failure) are being investigated as a complementary part of the safety concept. While the safety assessment and supporting studies are still ongoing, the results thus far indicate that the reference site (i.e., the Gorleben salt dome) is expected to maintain its integrity as a geologic barrier during the entire reference period of 1,000,000 yr.

C-3.2 Clay

C-3.2.1 Generic Clay Study (United States)

Sandia National Laboratories recently evaluated the feasibility of permanent disposal of high-activity nuclear waste in clay formations within the United States (Hansen et al. 2010, Section 1). In this context, clay refers to a group of related lithologies—mudstone, clay, shale, and argillite—having in common positive qualities for potential repository development and waste isolation. The Sandia study built upon the efforts of the international community to establish functional and operational requirements for disposal of a variety of waste forms in clay. Scoping performance analyses were performed based on the applicable FEPs identified by investigators from a number of countries.

Generic assumptions regarding repository geometry, material properties, thermal loading, and the like were made to represent a plausible reference repository in clay. The total depth chosen for the reference repository is 450 m, which is below the estimated depth of 300 m needed to avoid the potential adverse effects of contact with shallow groundwater circulation as well as long-term surface erosion (Hansen et al. 2010, Sections 2 and 4). Above the repository is 150 m of clay, then 100 m of a sandstone aquifer, and finally 200 m of sediments. Similar to European disposal concepts, the reference design calls for emplacement boreholes drilled horizontally from a horizontal access tunnel.

Multi-physics calculations provided insights as to the sensitive aspects of the underground setting. Using three-dimensional grids, the study team simulated the coupled thermal-hydrologic-chemical-mechanical behavior of the reference system. The selected level of thermal loading produced waste package temperatures approaching 100°C. Based on the results of the study, the team concluded that a clay repository could accept all waste from the current inventory for emplacement, with the use of up to 50 years of decay storage for the hottest used nuclear fuel. For the purposes of the safety assessment calculations, the radionuclide inventory was assumed to be from used nuclear fuel, which when compared to high-level radioactive waste has a greater number of radionuclides that could contribute to exposure. Retrievability, while not precluded, was not a design priority. Instead, the study emphasis was on evaluating the ability of a clay repository to provide for permanent disposal and isolation of the waste from the biosphere.

Three scenarios were developed regarding how radionuclides might travel from the repository to a hypothetical aquifer and then to the biosphere: (1) short-term, advective transport through the repository openings or the disturbed rock zone and up the shafts; (2) long-term, diffusive transport through the host clay upward from the emplacement boreholes; and (3) a stylized human intrusion scenario (Hansen et al. 2010, Section 3). The first scenario was not evaluated because of its short-term nature, the likely effectiveness of engineered seals, and the lack of a strong hydraulic pressure gradient to drive water through the repository and up the shafts. The third scenario also was not evaluated because it is stylized and only consequences are evaluated. Only the second scenario was considered in the generic safety assessment, using a one-dimensional advective-dispersive model formulation.

Under the nominal conditions of the second scenario, the initial postclosure period will be marked by increased temperatures caused by the heat-generating emplaced waste packages (Hansen et al. 2010, Section 5.1). Thermal-hydrologic-mechanical calculations indicate that the temperatures near the waste packages can be maintained below boiling and will decay to within a few degrees of the ambient temperature within a few decades or longer depending on the waste form. The thermal pulse, construction effects, and ventilation will cause clay dehydration and deformation resulting in a DRZ) extending a few meters out from the repository. However, clay formations commonly exhibit plastic deformation. Within a few centuries after waste emplacement, overburden pressures will tend to seal the fractures and resaturate the dehydrated rock.

The geologic barrier plays a significant role in limiting and/or delaying radionuclide transport. Radionuclides that do enter the clay host rock travel very slowly by diffusion because of the extremely low permeability. Advective transport under reasonable hydraulic gradients is insignificant over typical regulatory timeframes. Moreover, chemically-reducing conditions will limit radionuclide solubility, and thus mobility. Sorption of many radionuclides onto clays with high specific surface area will also retard transport. Because of the long transport times, most of the mobile radionuclides will decay before they can reach the biosphere.

The calculated dose to the reasonably maximally exposed individual, based on the radionuclide mass flux into a hypothetical overlying sandstone aquifer, is 0.01 mrem/yr or less at 1,000,000 years. The performance analysis predicts that the dose at 10,000 years is effectively zero. These results included conservative assumptions such as instantaneously degraded waste and waste forms, unlimited availability of moisture for waste form degradation and transport, and no sorption on degraded waste package materials.

Relevant findings of the study are summarized as follows:

- Clay mineralogy and chemistry combine to limit radionuclide transport through the influence of low permeability, chemically-reducing environments, and sorption.
- The clay repository concept will effectively isolate all the types of used nuclear fuel and high-level radioactive waste that currently exist in the U.S. inventory, considering thermal output, radiological characteristics, and transport to the biosphere.
- Generic performance analysis for a clay repository exhibits excellent performance.
- Experience with seal systems for the WIPP repository would provide significant support to design, construction, testing, and safety assessment for a clay repository.
- Coupled hydrogeochemical transport calculations indicate maximum extents of radionuclide transport on the order of tens to hundreds of meters, or less, in a million years. Under the conditions modeled, a clay repository could achieve total containment, with no releases to the environment in undisturbed scenarios.

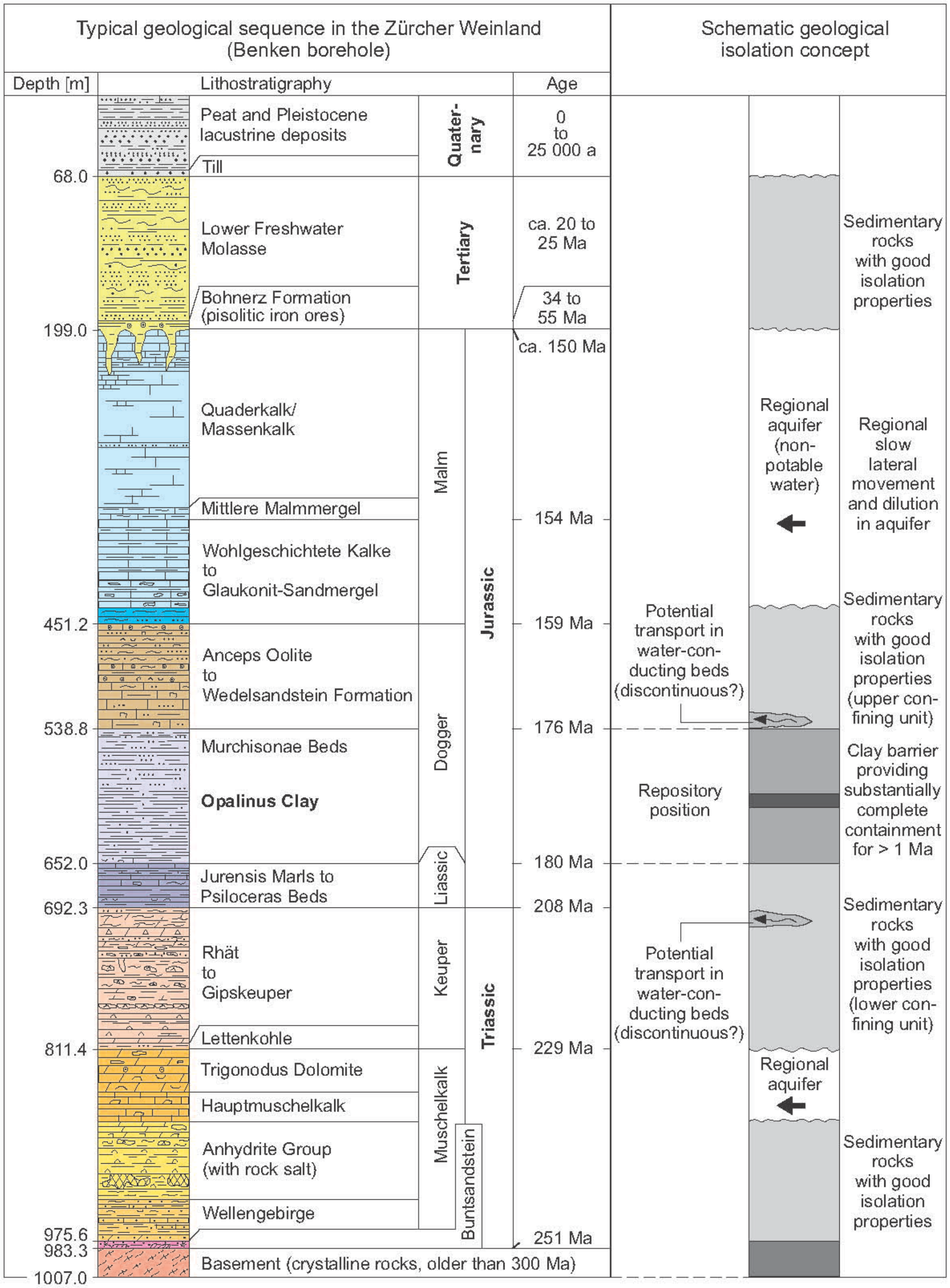
These findings lead to the conclusion that clay media are highly viable to host repositories for used nuclear fuel and high-level radioactive waste in the United States.

C-3.2.2 Opalinus Clay Safety Case (Switzerland)

In Switzerland, the responsibility for the safe management and disposal of radioactive waste lies with the producers of that waste (Nagra 2002, Section 1). The Swiss Federal Nuclear Safety Inspectorate (HSK) has supervisory authority. To help fulfill their obligations, the electrical utilities operating nuclear power plants and the Swiss government, which is responsible for management of medical radioactive waste, created the National Cooperative for the Disposal of Radioactive Waste (Nagra) in 1972. Over the years, Nagra has been conducting research and developing concepts for deep geologic disposal of radioactive waste. Northern Switzerland became the focus of extensive regional field investigation in the early 1980s, and a 1994 review confirmed the Opalinus Clay of the Zürcher Weinland region as the favored option. In 2002, Nagra presented its safety case for the disposal of spent fuel, high-level radioactive waste, and various types of intermediate-level radioactive waste in the Opalinus Clay.

The potential repository host rock is a well-compacted, moderately over-consolidated claystone (clay-shale) composed of the Opalinus Clay and other beds within the Opalinus Clay facies (Nagra 2002, Section 4). The sedimentary layers were created 180 million years ago when fine clay, quartz, and carbonate particles were deposited in a shallow marine environment. Located in the Molasse Basin, the Opalinus Clay is part of a thick sequence of Mesozoic and Tertiary sediments underlain by Paleozoic sediments and crystalline basement rocks. The overlying Tertiary sediments thicken considerably to the south. The Mesozoic sediments containing the Opalinus Clay are of uniform thickness over an area of several kilometers, almost flat-lying and little affected by faulting. To the north-east and in the Jura mountains to the west, the sedimentary rocks become faulted and folded. The region is consequently structurally simple and, on the northern edge of a deep basin, bounded by deformed sedimentary rocks to the north-east and west. The area is tectonically stable on a timescale of the next few million years, with a low rate of uplift and associated erosion and average in-situ stresses and heat flows.

The location of the host rock and the potential repository within a geological sequence from one of the research boreholes is illustrated in Figure C-3. The figure also provides a schematic representing the associated safety features contributing to the isolation concept. The host rock is a little over 100-m thick and has very low hydraulic conductivity. Thick (100- to 150-m), clay-rich formations above and below the host rock serve as upper and lower confining units respectively. Because they contain thin, probably discontinuous sandstones and carbonate rocks, they are not as impermeable and homogeneous as the Opalinus Clay. Yet, they still possess good isolation properties, with limited groundwater movement and good sorption potential. Above and below these confining units, the sequence contains large, regional aquifers in the carbonate rocks. However, neither of these aquifers or the permeable horizons is exploited in this region. The shallower regional aquifer has relatively high salinities making it nonpotable.



Source: Nagra 2002, Figure 4.2-7.

Figure C-3. The Geological Sequence (left) and the Simplified Features Illustrating the Isolation Concept (right) for the Opalinus Clay

The concept for the repository itself calls for a series of waste emplacement tunnels to be constructed at a depth of about 650 m, which is roughly mid-plane of the host rock (Figure C-3). The fact that the Opalinus Clay is an indurated claystone with good engineering properties allows for the construction of small, unlined tunnels and larger, lined tunnels at a reasonable cost. The design provides for disposal of three types of waste: spent fuel assemblies containing UO_2 or mixed oxide fuel; vitrified high-level waste from the reprocessing of spent fuel; and long-lived, intermediate-level waste. Horizontal rather than vertical emplacement was chosen to maximize the length of the radionuclide transport path to adjacent formations. After the waste is emplaced, the emplacement tunnels will be sealed. Before closure, the single shaft and the operations and construction tunnels will be backfilled with a bentonite-sand mixture and sealed with bentonite seals.

The natural and engineered systems contain multiple barrier features providing a variety of safety functions, some of which were illustrated in Figure C-3. The safety case recognizes six barrier features (Nagra 2002, Sections 6 and 9). Described below, the six features ensure that adequate levels of safety are provided for all realistically conceivable possibilities for the characteristics and evolution of the system:

1. **Deep Underground Location of the Repository**—This setting makes inadvertent human intrusion unlikely. Moreover, it is not disposed to disruptive geological events or processes unfavorable to long-term stability.
2. **Host Rock—**The Opalinus Clay has low hydraulic conductivity; fine, homogeneous pore structure; and self-sealing capacity. As such, flow is dominated by diffusion rather than advection providing a strong barrier to radionuclide transport.
3. **Chemical Environment—**The chemical environment promotes a range of geochemical processes favoring the immobilization and retardation of radionuclides. It also favors the long-term stability of the engineered barriers. The stability of the chemical environment is expected to be maintained by a variety of chemical buffering reactions.
4. **Bentonite Buffer (for spent fuel and high-level waste)**—The buffer provides a suitable environment for the canisters and waste forms. It is a well-defined interface between the canisters and the host rock that minimizes the effects of the existence of tunnels and heat-producing waste on the host rock. With properties similar to the host rock, it can provide a strong barrier to radionuclide transport.
5. **Spent Fuel and High-level Waste**—They provide stability in the expected environment and continue to retain most radionuclides after a canister breach.
6. **Spent Fuel and High-level Waste Canisters**—Because they are mechanically strong and corrosion resistant in the expected environment, the canisters can provide absolute containment for an extended time period.

Over time, Nagra has developed an understanding of the behavior of individual safety barriers from a number of sources including laboratory and field experiments and general observations from nature. To better understand how the barriers work together as a system, Nagra has analyzed a wide variety of safety assessment cases using both deterministic and probabilistic approaches. These cases can be organized into categories as follows: (1) the Reference Scenario, which examines the situation in which all six barrier features are assumed to operate as expected, (2) Alternative Scenario 1, which examines the release of volatile radionuclides along gas pathways, (3) Alternative Scenario 2, which examines the release of radionuclides affected by human actions, (4) “what if?” cases to investigate the robustness of the disposal system, (5) cases investigating design and system options, and (6) cases illustrating the effects of biosphere uncertainty.

Evaluation of the Reference Scenario began with simulating a reference case. Then additional cases were run to examine the uncertainty and variability of input parameter values used in the reference case. The results for the reference case show that most of the radionuclides decay within the spent fuel and high-level waste and the surrounding bentonite buffer, or for the intermediate-level waste, the cementitious buffer. The clay-rich confining units surrounding the host rock have good sorption properties and can also retard transport. Of the radionuclides that are released from the waste forms and buffer material, only the most mobile and long-lived radionuclides can reach the edge of the host rock formation given the time required for diffusive-controlled transport. Long-lived radionuclides that escape the confining units may enter the regional aquifers. However, those radionuclides are significantly dispersed and diluted. Additional dilution occurs when the deep aquifers discharge to the more dynamic freshwater flow systems of near-surface gravel aquifers or to river waters. In short, radionuclides that reach the biosphere do so in concentrations are too low to give rise to any safety concerns.

The maximum annual dose rate for all waste forms in the reference case is 5.3×10^{-5} mSv/yr (5.3×10^{-3} mrem/yr) occurring at about 1,000,000 years (Nagra 2002, Section 8). This calculated dose rate is well below the regulatory guideline of 0.1 mSv/yr (10 mrem/yr) and the typical natural background radiation in Switzerland of 1 to 20 mSv/yr (100 to 2000 mrem/yr). Twenty one other cases run as part of the Reference Scenario also yielded maximum annual dose rates that were several orders of magnitude below the regulatory guideline.

A total of 49 cases were run for the other two scenarios, the “what if?” cases, the design and systems operations cases, and the cases illustrating the effects of uncertainty. The maximum annual dose rates calculated for these cases were all at least an order of magnitude below the regulatory guideline, with most being several orders of magnitude below the regulatory guideline.

The findings of the safety assessment along with the multiple lines of evidence presented in the safety case support the conclusion that a deep geologic repository sited in the Opalinus Clay can provide for safe disposal of radioactive waste. An understanding of the effectiveness and reliability of the barrier features provides confidence in the robustness of the disposal system.

C-3.2.3 Dossier 2005 for Clay (France)

In 1991, the French National Radioactive Waste Management Agency (Andra) was tasked with assessing the feasibility of the deep geologic disposal for long-lived high-level waste while taking reversibility into consideration (Andra 2005a, Section 1). Although Andra evaluated both clay and granite as potential repository host rock types, the emphasis has been on clay.

After nearly 15 years of research on clay formations, Andra completed and submitted Dossier 2005 Argile—a series of five reference knowledge reports—to French authorities in 2005 (Andra 2005a, Section 2). These five reports contain information respectively on the geologic medium and the biosphere, the materials, the radioactive substances, waste behavior in the repository, and the inventory of long-lived high-level waste already produced or expected to be produced.

The geologic setting in Dossier 2005 Argile is the Paris basin in the sector south of the Meuse and north of the Haute-Marne. The basin consists of horizontal layers of limestone, marl, and clay deposited in ancient oceans. Of particular interest is the Callovo-Oxfordian layer, which was deposited about 155 million years ago. At least 130-m thick and located at a depth of between 400 and 600 m, this clay layer has been studied extensively using the Meuse/Haute-Marne underground research laboratory, which itself is located at a depth of about 490 m. Andra has also conducted experiments at the Mont Terri laboratory in Switzerland, thereby taking advantage of the fact that the Mont Terri argillites are quite similar to the Callovo-Oxfordian argillites.

Ten years of research at the Meuse/Haute-Marne site have confirmed that the Callovo-Oxfordian argillites have favorable properties with respect to deep geologic disposal. The clay layer is thick, regular, and homogeneous over a large surface area. The argillites provide a stable chemical environment and have the ability to retain a large amount of chemical elements. The primary flow and transport mechanism is diffusion. Water flow in the low permeability argillites is so slow that a drop of water only moves a few centimeters in 100,000 years. Water flow in the layers surrounding the Callovo-Oxfordian is also quite slow. While there is an aquifer located beneath the underground laboratory, it lacks the necessary properties to be of interest in the future as an exploitable resource either for consumption or for geothermal purposes. The seismic risk is very low given that tectonic deformation in the last 150 million years has been limited to two faults on the boundary of the sector studied. No faults have been found within the study sector itself.

Argillite exhibits good mechanical resistance, making it suitable for mining excavation work. The design calls for a mined repository on a single level in the middle of the clay layer. The engineered structures are also designed to minimize mechanical disturbances. The modular architecture is organized into distinct zones corresponding to waste package type: intermediate-level, long-lived waste (B waste); high-level, long-lived waste encased in a glass matrix (C waste); and nonprocessed spent fuels (CU waste), included for completeness of design but not currently planned for disposal.

Because the C and CU wastes give off a large amount of heat, the zones for these wastes are designed to provide for sufficient spacing between disposal cells and appropriate package arrangement to limit potential heat-related disturbances. To limit the potential for flow, the disposal cells are arranged in a dead-end manner and can be sealed with low permeability, swelling clay plugs if desired. As part of the preparations for closure, drifts and shafts will be sealed and backfilled. The engineered materials in the repository (e.g., cement, concrete, metal) have been selected not only for functionality, but also to minimize the potential for adverse impacts. As a result, the impacts are small and limited to the immediate surroundings of the engineered structures.

While reversible by design, the repository obviously must also be able to evolve safely over a long period of time without human intervention. The safety assessment documented in Dossier 2005 Argile examines the repository evolution under various scenarios over a time period of 1,000,000 years to ensure that the environmental impact is very low. The safety assessment included a normal evolution scenario as well as multiple altered evolution scenarios.

Under the normal evolution scenario, the thermal pulse experienced after repository closure has little effect on radionuclide release and transport. The design limits temperatures at all points to 90°C. The maximum temperatures are reached after one or more tens of years, but are expected to return to about 40°C by about 1000 years for the C waste and 6000 years for the CU waste. For either waste type, these durations are short compared to the time needed to deteriorate the waste packages. In addition, the thermal pulse does adversely impact the mineral composition of the host rock or its confinement capabilities.

The hydraulic evolution of the system starts with repository construction. The resulting disturbances to the initial hydraulic equilibrium are confined to the repository and the Callovo-Oxfordian layer. A new equilibrium is achieved between 100,000 and 200,000 years. After a few hundred thousand years to 1,000,000 years, the direction of flow in the layers surrounding the Callovo-Oxfordian is expected to change due to climatic changes and erosion. However, the flow velocity will remain very low. Erosion will not affect the Callovo-Oxfordian layer because it is too deep.

The construction of the repository also causes a disturbed rock zone extending a few meters into the host rock. This zone contains microfissures and depending on depth may also include fractures. Calculations indicate that no fracturing occurs at depths of 500 m, while at 600 m depth, fractures are moderately initiated. Although the argillites are stiff, they do deform slowly. As a result, the microfissures and fractures tend to heal on a timescale of several thousands to tens of thousands of years.

Three barriers act to prevent or delay the release and transport of radionuclides:

- **Waste Packages**—C and CU waste packages are expected to maintain their integrity until 4000 and 10,000 years respectively. Afterwards, water contacting the glass and spent fuel assemblies causes them to dissolve for several hundreds of thousands of years. For the B waste packages, releases due to water are expected to occur over several tens of thousands of years. This estimate for B waste neglects the several tens of thousands of years needed for degradation of the concrete overpacks.
- **Disposal Cells**—Travel by diffusion through the disposal cells to the Callovo-Oxfordian layer will take at least 100,000 years. In that time, most of the radionuclides will disappear through decay.
- **Geologic Medium**—Once in the Callovo-Oxfordian layer, most of the radionuclides will become trapped by one of two mechanisms. First, large amounts of smectite will tend to immobilize dissolved species. Second, the chemistry of the interstitial water causes some radionuclides to precipitate. The radionuclides that do not become trapped will migrate very slowly by diffusion. Only the most soluble, long-lived radionuclides will be able to reach the layers above and below the Callovo-Oxfordian during the next million years.

The safety assessment found that there is no significant impact to man and the environment for the normal evolution scenario. The safety margin included the effects of a number of cautious choices such as using pessimistic parameter values, assumptions, and conceptual models.

In the altered evolution scenarios, Andra explored the potential impacts of low probability events and processes such as seal and plug failure, defective C and CU waste containers, and borehole penetration of the repository. The expected dose for each scenario remained well below the limit of 0.25 mSv/yr (25 mrem/yr) set by the Basic Safety rule. Even given the extreme situation in which all safety functions were assumed to be severely degraded, the expected dose was still less than the limit.

The safety assessment and supporting scientific investigations documented in Dossier 2005 Argile demonstrate that the Callovo-Oxfordian argillites offer a robust and efficient means for safely disposing of high-level, long-lived radioactive waste.

C-3.2.4 SAFIR 2 (Belgium)

In Belgium, research on the long-term management of high-level and/or long-lived waste began as early as 1974 (ONDRAF/NIRAS 2001, Section 1). Conducted by the Belgian Nuclear Research Centre (SCK•CEN), the investigation was quickly directed towards disposal in a stable geological formation with suitable properties. When argillaceous rocks were identified as the only such suitable medium in Belgium, efforts were focused on examining the Boom Clay layer beneath the Mol–Dessel nuclear zone at the SCK•CEN site. While SCK•CEN still conducts research, a single entity currently known as the Belgian Agency for Radioactive Waste and Enriched Fissile Materials (ONDRAF/NIRAS) was created in 1980 to manage radioactive waste in Belgium.

ONDRAF/NIRAS has prepared two safety reports documenting the first and second phases of methodological research and development. The first phase (1974 to 1989) was addressed in The *Safety Assessment and Feasibility Interim Report* (SAFIR). Conducted from 1990 to 2000, the second phase was documented in *Safety Assessment and Feasibility Interim Report 2* (SAFIR 2). SAFIR 2 provides an assessment of the confidence gained in the safety, feasibility, and robustness of waste disposal in the reference system. While the details of the generic repository concept have continued to evolve over time (EPRI 2010, Section 2), the discussion below focuses on the safety assessment and supporting information found in SAFIR 2.

The reference host formation selected for the purposes of evaluating a deep disposal solution is the Boom Clay (ONDRAF/NIRAS 2001, Section 3.2). The corresponding reference site is the nuclear zone of Mol–Dresser. While this reference system has been the primary subject of research, there has also been some preliminary study of an alternative reference host formation and site, i.e., the Ypresian Clays under the nuclear zone of Doel.

The Boom Clay is a poorly indurated clay with silty bands that are several tens of centimeters thick (ONDRAF/NIRAS 2001, Section 3.2.2). Early in the Belgian program, not much was known, nationally or internationally, about excavating underground facilities at a depth of some 200 m in a geologic medium with poor consolidation (ONDRAF/NIRAS 2001, Section 1). As a result, an underground facility—the High-Activity Disposal Experimental Site (HADES)—was constructed as a resource to help assess and demonstrate the feasibility of operating in this type of medium.

The Boom Clay lies in the northeast of Belgium within the Campine Basin, a sedimentary basin with quasi-horizontal strata. The Boom Clay itself dates from the Rupelian, a part of the Tertiary period lasting from 36 to 30 million years ago (ONDRAF/NIRAS 2001, Section 3.2.2). There are three members: Belsele-Waas (bottom), Terhagen (middle), and Putte (top). The total porosity of the Boom Clay is about 30% to 40% by volume. The bulk of the water is not free, but rather bound to the clay minerals. However, a sequence of aquifers and aquitards exist above and below the Boom Clay (ONDRAF/NIRAS 2001, Section 3.2.3). The most notable—the Neogene Aquifer—is located in the sands overlying the Boom Clay. Because of its substantial thickness, high hydraulic conductivity, and low salinity, the aquifer is an important source of drinking water.

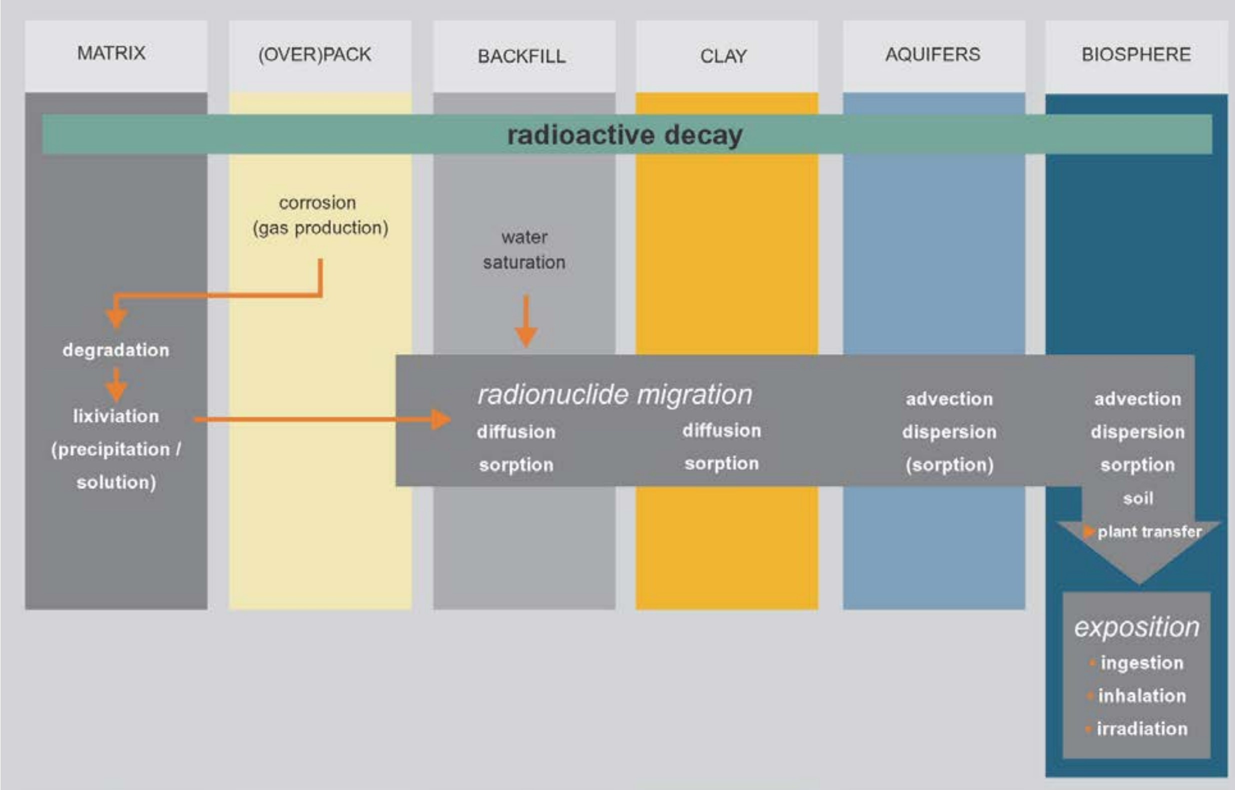
The Boom Clay possesses a number of characteristics that enhance its potential as a geologic barrier to radionuclide transport. The vertical hydraulic conductivity is very low as is the natural hydraulic gradient (ONDRAF/NIRAS 2001, Section 3.5.1). As a result, the dominant flow mechanism is diffusion, which means that radionuclide transport would be controlled by concentration gradients and would be very slow. There are also several geochemical and physic-chemical characteristics that promote radionuclide retention. The formation is a reducing and slightly alkaline medium, a condition that supports the reduction of species sensitive to redox potential to species of low solubility. It also has a high capacity for cation exchange, and it acts as an ultrafilter for colloids.

The seismic activity level in the Campine Basin is low (ONDRAF/NIRAS 2001, Section 3.2.2.2). The basin generally lacks large-scale structural features with the exception of the Roermond Graben in the northeastern part of the basin. The Roermond Graben is an active rift valley causing subsidence in the valley center of about 0.25 to 0.5 mm/yr over the last 2,000,000 years. Research regarding the potential for seismic activity in the future is ongoing.

The types of waste addressed in SAFIR 2 are Category B waste (long lived, low heat output) and Category C waste (highly radioactive, mostly long lived, moderate to high heat output) (ONDRAF/NIRAS 2001, Sections 3.1 and 3.1.1). Two options are considered: an option in which all spent fuel types are reprocessed, and an option in which there is no reprocessing before disposal (ONDRAF/NIRAS 2001, Sections 3.1.2 and 3.3).

The repository layout consists of a system of rectilinear galleries located in the median plane of the Boom Clay layer, approximately 240 m below ground level (ONDRAF/NIRAS 2001, Section 3.3). Two shafts provide access to the gallery system. The disposal galleries for vitrified waste are on one side of a connecting gallery, and those for other waste are on the other side. At closure, the repository will be backfilled with a mixture of sand and a swelling clay-based material and the main galleries, connecting gallery, and access shafts will be sealed.

The safety assessment calculations in SAFIR 2 emphasized the normal evolution scenario in which the natural and engineered barriers were assumed to perform as expected (ONDRAF/NIRAS 2001, Section 4.1.2). Figure C-4 illustrates the main components of the disposal system and the surrounding environment as well as the main processes included in this scenario.



Source: ONDRAF/NIRAS 2001, Figure 4.3.

Figure C-4. Components and Processes Considered in the Normal Evolution Scenario for SAFIR 2

In the normal evolution scenario, waste packages and backfill materials are expected to degrade over time (ONDRAF/NIRAS 2001, Section 4.2). The waste matrices will continue to function for a few hundred years to several tens of thousands or even several hundreds of thousands of years, with the longer durations belonging to the glass and uranium oxide matrices. The packagings and overpacks will last for a thousand years to several tens of thousands of years. The backfill, a clay-based material, will maintain its integrity for several thousands of years, in large part because the design avoids exposure to excessive temperature increases. While some radionuclides will be sorbed by the backfill material, it is the Boom Clay that plays the dominant role in terms of preventing or delaying radionuclide transport. Migration through the Boom Clay occurs primarily through the slow process of molecular diffusion. Sorption by the clay minerals or by organic materials in the clay will halt the transport of many radionuclides. The few radionuclides that migrate through the clay layer will be diluted in the overlying aquifers, where advection and dispersion dominate. Some sorption is also possible because of minerals present in the aquifers. Mobile radionuclides in the aquifers may eventually be transported to the biosphere.

Besides the normal evolution scenario, multiple altered evolution scenarios were considered in a series of qualitative and quantitative assessments (ONDRAF/NIRAS 2001, Section 4.2). The altered evolution scenarios are based on the assumption that one or more barriers partially or completely fails to perform its intended function. The scenarios investigated include the following: resource exploitation drilling, anthropogenic climate change, fault activation, severe glaciation, inadequate sealing, and premature failure of an engineered barrier.

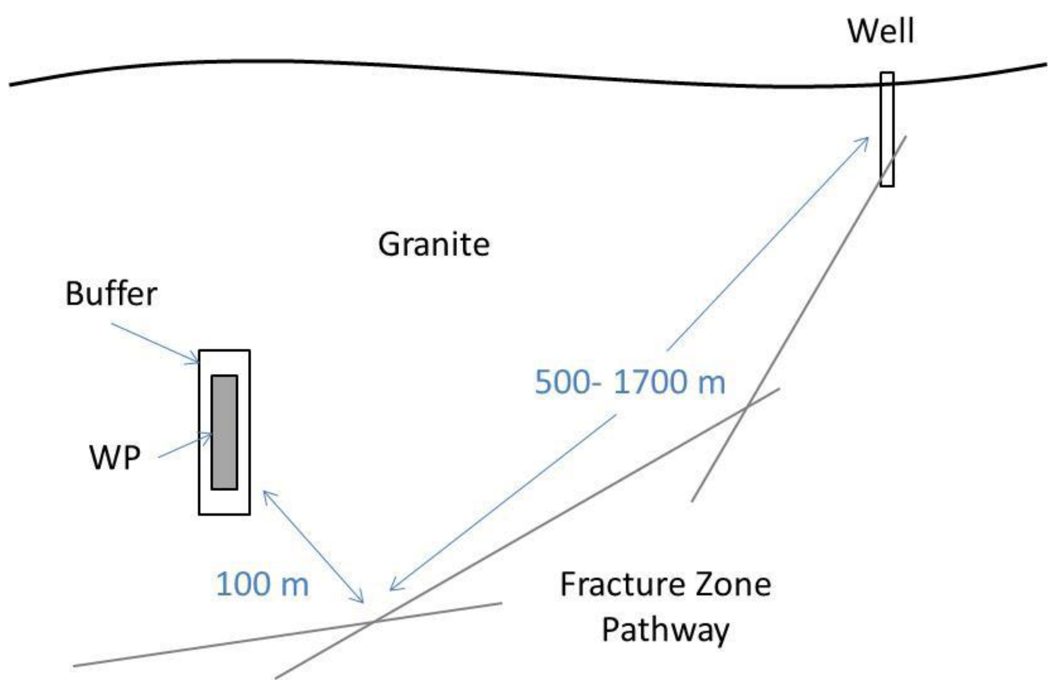
The results of all scenarios, whether normal or altered evolution, confirm that the dominant role played by the clay layer acting as a geologic barrier preventing or delaying radionuclide transport (ONDRAF/NIRAS 2001, Section 6.3). The calculations for the normal evolution scenario indicate that the exposure for an individual in the reference group is systematically and clearly below the dose limit. The initial results for the altered evolution scenarios highlight the robustness of the reference disposal system. The overall performance of the system remained broadly intact despite the various assumptions compromising system integrity.

C-3.3 Granite

C-3.3.1 Generic Granite Study (United States)

The United States has many granite formations with positive attributes for the potential disposal of high-activity waste. In 2011, Sandia National Laboratories conducted a study to evaluate the feasibility of deep geologic disposal using granite as the host rock (Mariner et al. 2011). The study drew from the knowledge base for functional and operational requirements generated by the international community, particularly Sweden and Finland. The study team considered applicable FEPs identified by investigators from a number of countries and developed multiple potential scenarios of repository evolution. For scoping purposes, a preliminary safety assessment of two of the scenarios was conducted to provide additional insight regarding postclosure behavior of a generic repository in granite.

Generic assumptions were made to construct a plausible representation of a reference repository in granite. The reference repository was assumed to be located at a depth of 500 m in a saturated granite formation that is reasonably homogeneous and sparsely fractured with low permeability (Mariner et al. 2011, Sections 2 and 4.1). By siting the repository at least 300-m below the surface, erosion and shallow groundwater circulation will not affect repository performance. The depth also allows the repository to maintain reducing conditions even during periods of deep penetration of glacial melt water. Construction of a future groundwater well is assumed to occur at a distance of 500-m down-gradient of the repository boundary. The conceptual model for transport under nominal conditions includes a pathway through a near-field fracture to a far-field fracture zone connecting with the groundwater well, as illustrated in Figure C-5.



Source: Mariner et al. 2011, Figure 4-1.

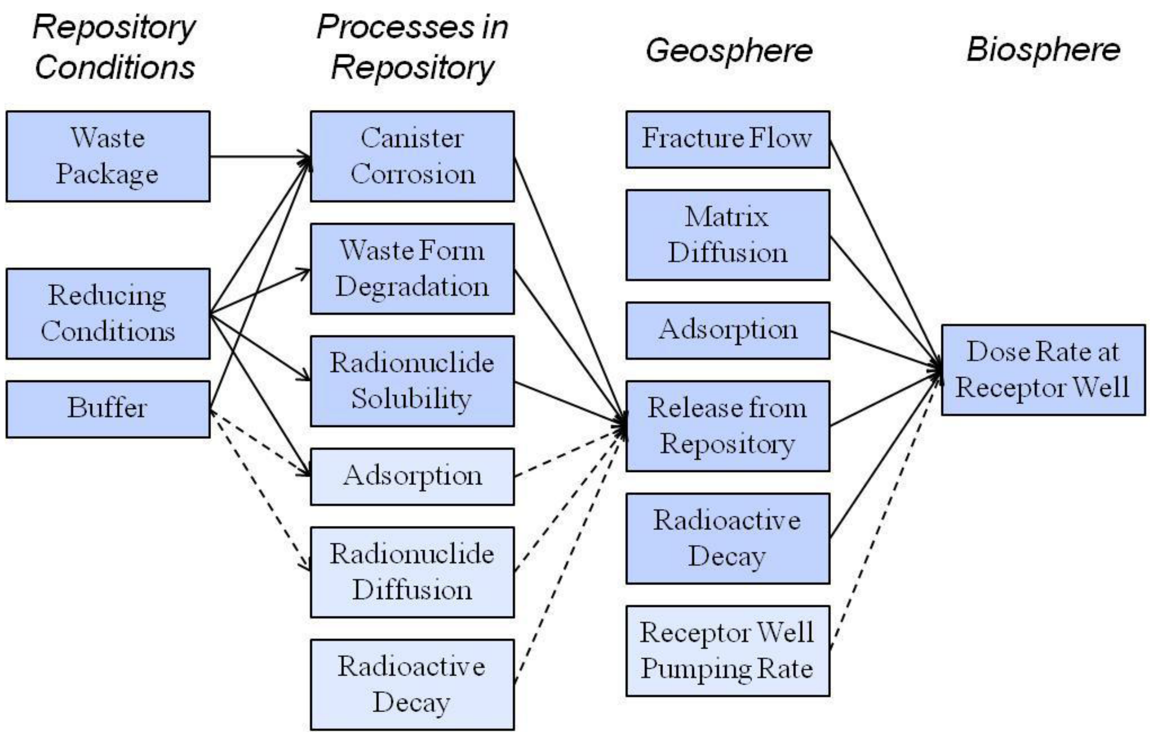
Figure C-5. Conceptual Model of the Radionuclide Pathway for the Nominal Scenario

The study considered an inventory consisting of used nuclear fuel, high-level radioactive waste glass, and other reprocessed waste. However, for the purposes of the preliminary safety assessments, the waste was assumed to be used nuclear fuel. The reference layout was based on disposal concepts in Sweden and Finland. It consists of a network of tunnels in which copper canisters containing the waste are emplaced in vertical boreholes drilled in the tunnel floor. Compared to the safety concepts for salt, clay, or deep boreholes, the safety concept for granite disposal relies more heavily on the engineered barrier design to preserve the waste packages. A clay buffer surrounds the waste package in the borehole, and the emplacement tunnels are backfilled with a mixture of crushed rock and clay. To prevent damage to the buffer, the design may need to prevent temperatures from exceeding 100°C. While retrievability is facilitated by the long-term stability of granite, the design priority is on the permanent disposal and isolation of the waste from the biosphere.

The study identified five potential scenarios based on analysis of the relevant FEPs as well as scenarios identified in various national programs in other countries (Mariner et al. 2011, Section 3.2):

- **Nominal Scenario**—The waste packages and engineered barrier system perform as designed. No releases occur because the waste form remains contained within the waste package during the entire performance period.
- **Defective Waste Package Scenario—**A major defect in one waste package at the time of emplacement allows early radionuclide release.
- **Buffer Failure Scenario—**Deep circulation of glacial melt waters erodes buffers exposing one fourth of the waste packages to advective groundwater flow. The increased corrosion causes a small fraction of these waste packages to fail within the performance period.
- **Shear Movement Scenario**—An earthquake causes a displacement that ruptures waste packages.
- **Disruptive, Human Intrusion Scenario—**A borehole is drilled through the repository and later abandoned; a vertical hydrologic gradient transports radionuclides to a shallow aquifer from which they are pumped to the biosphere.

The preliminary safety assessment addressed two of the scenarios: the defective waste package scenario (deterministic treatment) and the buffer failure scenario (probabilistic treatment) (Mariner et al. 2011, Section 4). Despite the use of some conservative assumptions, the calculated dose rates at the hypothetical accessible environment for both scenarios are well below acceptable safety limits. A qualitative evaluation of the relative importance of the modeled FEPs was performed by visually comparing calculated radionuclide masses within the waste package, buffer, near-field, and far-field domains over time and releases from these domains over time. The FEPs that were deemed to be of primary and secondary importance are summarized in Figure C-6. The darker blue boxes indicate highly important features and processes, and the lighter boxes indicate features and processes of intermediate importance.



NOTE: The darker blue boxes indicate highly important features and processes. The lighter boxes indicate features and processes of intermediate importance.

Source: Mariner et al. 2011, Figure 4-15.

Figure C-6. Relative Importance of Features and Processes Based on Qualitative Evaluation of Safety Assessment Results

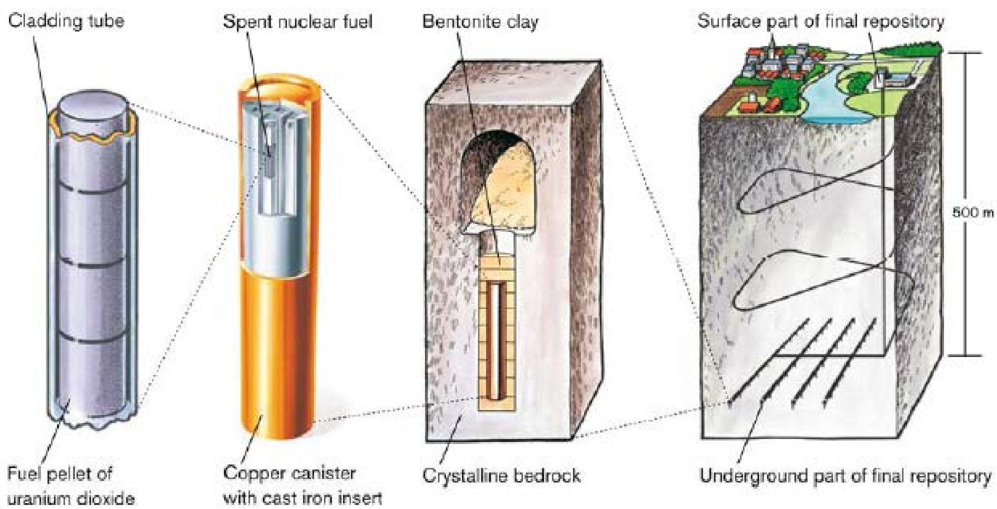
The most important simulated processes for preventing release of radionuclides from the repository are canister corrosion, waste form degradation, and radionuclide precipitation. These processes, in turn, depend highly on reducing conditions and the presence and properties of the canister and buffer. The buffer, in addition to delaying waste package failure, presents a diffusive and sorptive barrier to radionuclide transport; however, the results suggest that the buffer's role in limiting canister corrosion rates is more important to the release of radionuclides to the geosphere. Once the radionuclides enter the geosphere, fracture flow velocities, matrix diffusion, sorption, and radioactive decay are of the highest importance to the dose rate at the receptor well. The importance of radioactive decay is magnified by the long residence times in the repository and geosphere compared to the relatively short half-lives of many of the radionuclides. In both scenarios, the dominant radionuclide contributing to dose at the receptor well is ^{129}I, which has a long half-life of 1.57×10^7 years.

The results of the preliminary safety assessment performed in this study are consistent with the results of the more comprehensive safety assessments performed for sites in Sweden, Finland, and Canada (Mariner et al. 2011, Section 5). The study concludes that it should be possible to construct a granite repository in the United States that would satisfy established safety criteria. It also indicates that a small number of FEPs would largely control the release and transport of radionuclides. These FEPs would be needed to inform a more comprehensive safety assessment.

C-3.3.2 Forsmark License Application (Sweden)

The Swedish have worked on research, development, and demonstration of deep geological disposal of spent fuel for more than 30 years (Thegerström and Olsson 2011). In 1973, the nuclear power utilities formed the Swedish Nuclear Fuel and Waste Management Company (SKB) with a primary mission to coordinate the nuclear fuel supply. The mission changed three years later when the Swedish government passed legislation making the nuclear power utilities responsible for the safe disposal of the spent fuel.

After much research and development, the SKB proposed the KBS-3 method as the preferred method of disposal (SKB 2011, Section 1). The KBS-3 method specifies that the spent fuel be contained inside copper canisters with a cast iron insert. The canisters are surrounded by bentonite clay in a repository located at a depth of about 500 m in crystalline bedrock, as seen in Figure C-7. Evaluation of potential alternatives for spent fuel disposal has confirmed that none have significant advantages over the KBS-3 method for the Swedish program (Thegerström and Olsson 2011).



Source: SKB 2011, Figure 1-1.

Figure C-7. The KBS-3 Concept for Final Storage of Spent Nuclear Fuel

Development of the KBS-3 concept has been guided by a number of principles that constitute the safety philosophy for the design of a final repository (SKB 2011, Section 2). Disposal is to be at depth in the Swedish crystalline bedrock, which is a long-term stable geological environment. The host rock must be such that there will be no economic interest by future generations. Multiple engineered and natural barriers will surround the spent fuel. Containing the spent fuel within the canister is the primary safety function. The secondary safety function of the barriers is to retard a potential release from the repository in the event of a container breach. Naturally occurring materials having long-term stability will be used for the engineered barriers. The design will avoid adverse impacts such as those from temperature and radiation on the barriers. The barriers are to be passive so that their functionality is maintained without human intervention or an artificial supply of matter or energy.

Even before the KBS-3 method was proposed, the effort to find a suitable site was underway (Thegerström and Olsson 2011). From 1977 to 1985, eight sites, called study sites, from all over Sweden were the subject of comprehensive investigations. The investigations yielded a great deal of data indicating that Sweden has many places with good geological characteristics for a potential repository. An important finding of the studies was that one could not attribute more or less suitability to a particular part of the country or a special geological environment within the crystalline bedrock. Rather, local conditions are the most important factor.

Between 1993 and 2000, the SKB conducted feasibility studies in eight municipalities to determine whether further siting studies were warranted for the municipality in question. At the same time, the municipality and its citizens were given an opportunity to form an opinion and consider the possibility of further participation without making any commitments.

By 2002, the search had narrowed to two potential sites: the Simpevarp and Laxemar areas (Oskarshamn municipality) and the Forsmark area (Östhammar municipality). From 2002 to 2008, SKB conducted investigations of rock characteristics that included measurements from the air, the surface, and in 1,000-m-deep boreholes. At each site, a target area of approximately 10 km^2 was investigated. To gain additional information, SKB drilled and cored about 20 boreholes to potential repository depth or deeper. In addition, SKB examined natural and cultural values focusing on the effects a repository might have on society. Based on a systematic examination of both sites, SKB selected Forsmark as the site for the final repository in 2009. Industrial considerations and environmental impacts were deemed similar for both sites. The primary advantage offered by Forsmark was that few water-conducting fractures exist in the rock at repository depth.

A part of the Fennoscandian Shield, the crystalline bedrock at Forsmark was formed between 1.89 and 1.85 billion years ago during the Svecokarelian orogeny (SKB 2011, Section 4). In some areas, called tectonic lenses, the effect of ductile deformation on the bedrock is limited, but there are adjacent ductile high-strain belts. The candidate area for the repository is located in one of these tectonic lenses. Below a depth of about 300 m, there are relatively few open or partly open fractures. Diffusion is the dominant transport mechanism in the rock matrix, which also has favorable sorption properties. A reducing chemical environment and salinity at repository depth help ensure the stability of the bentonite buffer.

The safety assessment, SR-Site, is a main component in SKB's license application to construct and operate a final repository for spent nuclear fuel at Forsmark (SKB 2011, Section 1). Its role in the application is to demonstrate long-term safety for a repository. The principal regulatory acceptance criterion corresponds to an effective dose limit of 1.4×10^{-2} mSv/yr (1.4 mrem/yr), which is about 1% of the natural background radiation (1 mSv/yr (100 mrem/yr)) in Sweden. Swedish regulations also imply that the assessment time for a repository of this type is one million years after closure. Findings from the Swedish regulator's review of an earlier version of the safety assessment have been taken into account in the SR-Site assessment.

The waste package is the principle barrier to radionuclide release in the safety assessment. An analysis of the containment potential suggests that two scenarios involving canister failure cannot be excluded. The first is the corrosion scenario and the second is the shear load scenario. For the corrosion scenario, investigators analyzed a series of cases representing a range of potential extents of corrosion failure. The resulting mean doses were at least one order of magnitude below the dose associated with the regulatory risk limit. In the most pessimistic cases, the first releases from canister failures occur after about 50,000 years. The calculated mean dose is about two orders of magnitude below the regulatory limit at 100,000 years and about one order of magnitude below the limit at 1,000,000 years.

For the shear load scenario, investigators used the containment potential analysis results to derive pessimistic values for the frequencies of shear load-induced canister failure. The probability-weighted consequences of shear failure of the canisters result in mean annual effective dose limits that are about three orders of below the regulatory limit between 1,000 and 100,000 years, before increasing to about two orders of magnitude below the regulatory limit at 1,000,000 years.

According to sensitivity studies, most of the uncertainty in the calculated dose comes from input uncertainties for the fuel dissolution rate, the failure time of the canister, and the flow-related transport resistance in the geosphere. Investigators addressed additional uncertainties by (1) forming variant calculation cases regarding, for example, different conceptual hydrogeological models, or (2) making pessimistic assumptions such as the likelihood of canister failure due to shear load.

The SR-Site safety assessment results, in conjunction with the wealth of supporting scientific knowledge gained over the years regarding the potential site and the KBS-3 repository concept, provides confidence that a deep geologic repository for nuclear waste can be constructed and safely operated at Forsmark.

C-3.3.3 Posiva's Preliminary Safety Case for the Olkiluoto Site (Finland)

In Finland, spent nuclear fuel as well as all other nuclear wastes that are generated within the country must also be processed, stored, and disposed of within the country (Posiva 2010a, Section 1). In 1995 the two Finnish nuclear power companies created Posiva Oy (Posiva) to conduct research and implement the final disposal program for spent nuclear fuel. Posiva later identified Olkiluoto as the site for a potential geologic repository based on a step-wise program of preliminary site characterization and safety assessment comparisons. Olkiluoto, a small island in the southwest of Finland, is already the host for two nuclear power reactors. In 2001, the Finnish parliament issued a Decision-in-Principle expressing its agreement with Posiva's recommendation.

In a 2003 decision, the Ministry of Trade and Industry directed Posiva to submit a license application by the end of 2012 for the construction of a disposal facility at Olkiluoto. An outline of the Preliminary Safety Analysis Report was submitted in 2009 and some preliminary safety assessments have been conducted. The Final Safety Analysis Report is scheduled for submittal in 2018 with the operational license application, and the goal for commencing disposal operations is 2020.

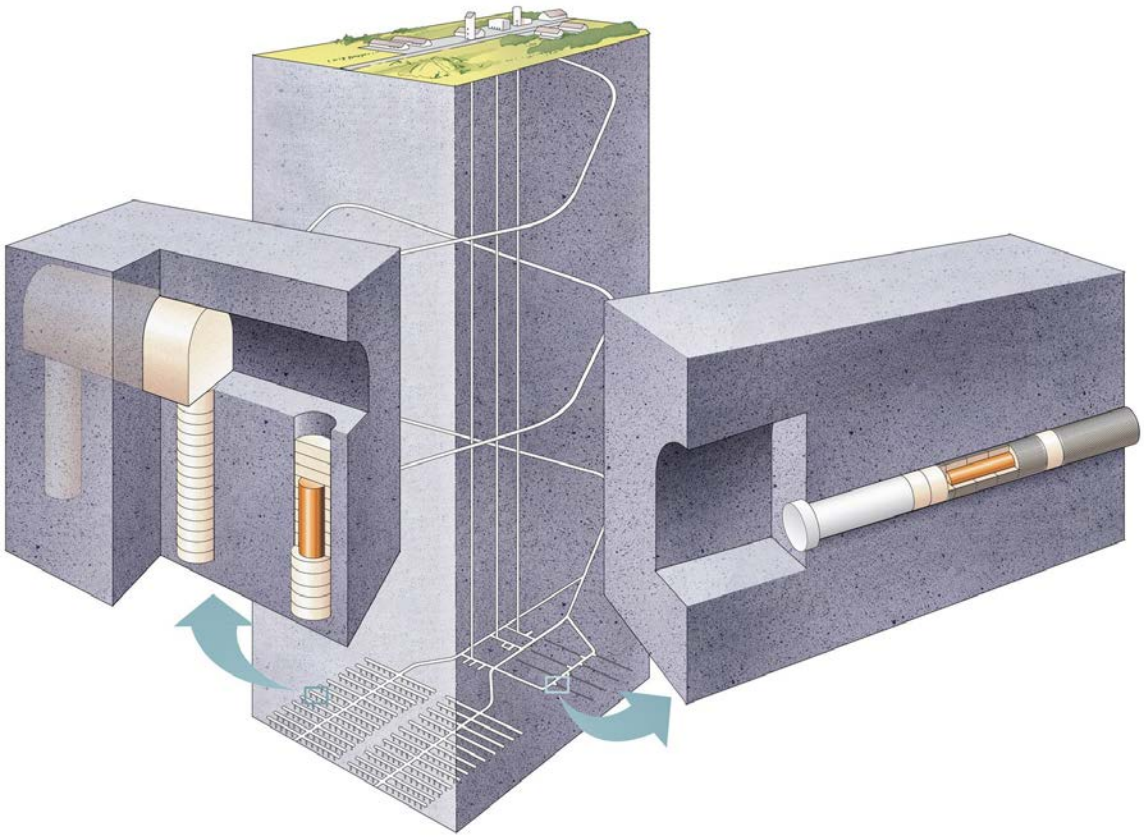
The Olkiluoto site has been studied for over 20 years (Posiva 2010a, Section 3.2). Site-specific underground investigations were made possible with the construction of the ONKALO underground research facility in 2004. The proposed location for the repository is near the center of the island, which has an area of about 12 km^2 and an average height of 5 m above sea level. The bedrock is of two types: metamorphic (mostly gneisses) and igneous (mostly granites). Because of the low matrix permeability, groundwater flow occurs primarily through fractures and fracture zones. The groundwater table is shallow, generally existing from the surface to 2 m below the surface. At repository depth, the more transmissive fractures tend to be located in local zones of abundant fracturing. Like the rest of the country, the island has little seismic activity.

The repository is envisioned as a one-level underground facility with tunnels at a depth of about 420 m (Posiva 2010a, Sections 1.1 and 3.3). It is being developed based on the KBS-3 concept originated by SKB in Sweden. As illustrated in Figure C-8, Posiva is investigating two variants:

- **KBS-3V**—Canisters will be put in deposition holes along the floor of long, horizontal deposition tunnels. Void spaces in deposition holes will be filled with bentonite buffer.
- **KBS-3H**—Canisters will be emplaced horizontally in 100- to 300-ft-long drifts. They are prepackaged into a supercontainer. Bentonite buffer separates canisters from bedrock and other canisters.

The canisters are mechanically strong, each one consisting of a cast iron insert surrounded by a copper overpack. The corrosion resistance of copper has been demonstrated by studies of natural and anthropogenic analogues. In preparation for closure, all void spaces in the repository will be backfilled and sealed, with possible exception of selected boreholes needed for remote monitoring. Retrievability is not part of plan.

For the planned 2012 construction license application, the reference design will be based on KBS-3V. The KBS-3H variant will be included as well, but with less rigor. A final decision regarding which variant to implement is expected during the preparations for construction in the 2015-to-2016 timeframe.



Source: Posiva 2010a, Figure 1-2.

Figure C-8. Schematic of the Two KBS-3 Variants Being Investigated by Posiva: KBS-3V (left) and KBS-3H (right)

Posiva considered the relevant FEPs and the associated uncertainty in developing possible scenarios for the evolution of the disposal system (Posiva 2010a, Sections 4 and 6). The resulting scenarios can be organized into three broad categories: climatic scenarios, which establish the framework for describing possible lines of evolution; the base scenario, which addresses the lines of evolution with no radionuclide releases; and assessment scenarios, which address the lines of evolution that include radionuclide release such as various modes of canister failure and human intrusion. Thus far, not all of the identified scenarios have been analyzed. For example, the human intrusion scenarios are among those being evaluated as part of the preparations for the planned 2012 license application.

The base scenario represents the expected evolution for most canisters in the repository over a period of at least hundreds of thousands of years (Posiva 2010a, Section 10). At closure, the bentonite clay buffer surrounding the canisters is only partly saturated with water, but it will become fully saturated as water moves from the rock into the buffer. The subsequent swelling will cause the buffer to fill any gaps between the canister and the rock, providing a protective environment. Corrosion by the oxygen available at closure will be limited as the oxygen is depleted by this and other chemical reactions. In addition, the saturated buffer will inhibit microbial activity. The flow of sulfide from the rock will also be inhibited; however, the small amounts that do reach the canisters will cause corrosion to occur at an extremely slow rate. The calculated time to canister failure due to such corrosion is on the order of millions of years. Changes at repository depth brought on by climate change will be more limited than those at the surface, and as a result the effect on canisters is not likely to be significant.

Although most of the canisters are expected to remain intact over at least several hundreds of thousands of years, the possibility that a small fraction of canisters could fail within this time period cannot be excluded. The key barriers that prevent or delay radionuclide transport and release are the following:

- Copper-iron canister, which after being breached could still provide some level of barrier performance for a period of time
- Bentonite buffer
- Backfill of the KBS-3V tunnels
- Host rock

The following features of these barriers contribute to the goal of limiting potential releases to the biosphere from a failed canister:

- Low groundwater flow rates
- Low dissolution rates of spent fuel under reducing conditions and low corrosion rates of fuel assembly materials
- Low solubilities of several of the most hazardous radionuclides
- Slow transport of radionuclides through the bentonite buffer (no contributions from advection or colloidal transport; sorption included)
- Slow transport of radionuclides through the host rock (limited groundwater flow, diffusion into rock matrix, sorption into rock matrix.

Finnish regulations specify a dose limit of 0.1 mSv/yr (10 mrem/yr) for the most exposed members of the public over the first several thousand years after closure. To put this criterion into perspective, the typical radiation exposure from natural sources is about 0.4 to 3 mSv/yr (40 to 300 mrem/yr), and the average exposure from natural and man-made sources in Finland is about 4 mSv/yr (400 mrem/yr). The calculated dose maxima were well below the regulatory limit. For example, the annual landscape dose maxima to a representative person for the most exposed group given a KBS-3V repository were below 10^{-4} mSv (10^{-2} mrem) in all cases. An analysis of the results for the KBS-3V and KBS-3H design variants indicated that the differences in geometry and transport paths have only minor impact on calculated releases and doses. Based on the results of the safety assessments and related studies conducted thus far, both variants show promise in terms of their ability to provide for long-term safe disposal of nuclear waste.

C-3.3.4 Third and Fourth Case Studies in Granite (Canada)

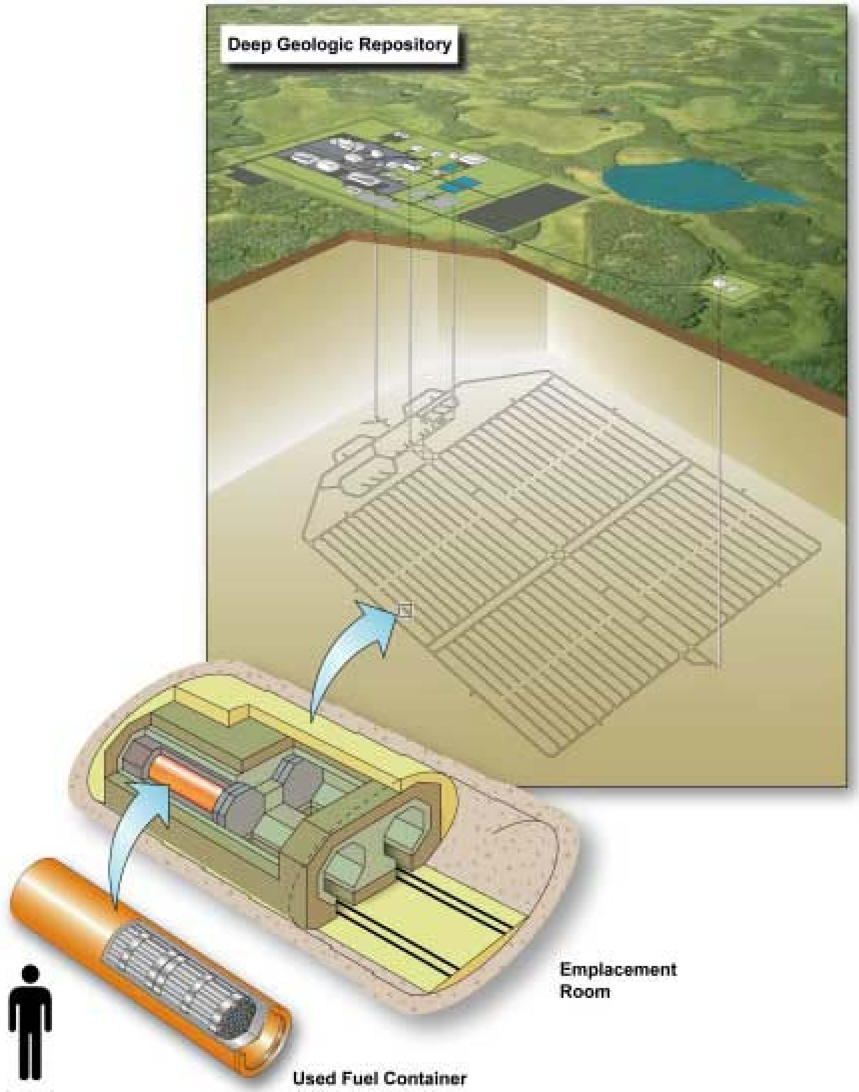
Canada has been investigating the options for the storage and disposal of spent nuclear fuel for about 30 years (Gierszewski et al. 2004, Section 1.1; Kremer et al. 2011). In 2002, the Canadian Nuclear Fuel Waste Act directed that the Nuclear Waste Management Organization be created to compare approaches, consult with the public, and recommend a long-term management approach.

Multiple safety assessments have been conducted to explore the behavior of a hypothetical repository located in the Canadian Shield rock. The effort began with the Environmental Impact Statement study presented in 1994. This study considered the vertical emplacement of titanium alloy containers in sparsely fractured granitic rock with low permeability. Presented in 1996, the Second Case Study considered the horizontal emplacement of copper containers in granitic rock with substantially higher permeability. This subsection focuses on the Third and Fourth Case Studies, which were presented in 2004 and 2011 respectively. Both safety assessments considered hypothetical repositories deemed to be more representative of potential sites that could exist within the Canadian Shield.

Third Case Study—For this safety assessment, the hypothetical repository was located in a fractured, granitic rock of intermediate permeability (Gierszewski et al. 2004, Sections 3.7 and 3.8). To better capture the behavior of a complex set of interconnecting major fractures and discontinuities, a geostatistical, discrete fracture network was generated based on Canadian Shield fracture statistics. The Third Case Study considers only one representation of the fracture network, but recognizes that a range of possible fracture networks could be considered during a site evaluation to address model uncertainty. Most of the hypothetical site is assumed to have granite extending close to the surface, with some sedimentary layers existing in low-lying areas such as near rivers and lakes. The water table is shallow, and there are no aquifers.

Located about 670 m below the surface, the repository is assumed not to intersect major fracture zones (Gierszewski et al. 2004, Section 3). Figure C-9 illustrates the layout of repository tunnels and rooms with exploded views of an emplacement room and a container. With this disposal concept, copper and steel containers are emplaced horizontally within vault rooms. Upon emplacement, the containers would be surrounded by layers of clay-based materials, with the inner-most layer being a bentonite buffer. At closure, access shafts and tunnels are backfilled and sealed. The safety assessment assumes the repository will be used to dispose of CANDU® (Canada Deuterium Uranium) spent fuel bundles from pressurized heavy water reactors designed in Canada. A representative spent fuel bundle is contains about 19.2 kg U (initial). Each container is designed to hold 324 bundles, or about 6.2 Mg U. CANDU® spent fuel comprises the bulk of the high-activity waste generated in Canada. Low- and intermediate-level wastes are not considered.

The Third Case Study provides an analysis of three scenarios: the Base Scenario, in which evolution occurs as expected; the Defective Container Scenario, in which an undetected defect exists in a few containers; and the Human Intrusion Scenario, in which future people unknowingly drill a borehole into the repository (Gierszewski et al. 2004, Section 4). The results are compared to the recommendations of the International Committee on Radiation Protection, ICRP 81 and to the average Canadian natural background radiation (Gierszewski et al. 2004, Section 2.2). For natural processes, ICRP 81 specifies a dose constraint to a member of the public of 0.3 mSv/yr (30 mrem/yr). The natural background radiation is assumed to be 1.7 mSv/yr.



NOTE: Human figure shown for scale.

Source: Gierszewski et al. 2004, Figure 3.1.

Figure C-9. Illustration of the Deep Geologic Repository Concept Showing the Container, Emplacement Room, and the Room and Tunnel Layout Considered in the Third Case Study

In the Base Scenario, the first 1,000 years after closure are marked by strong physical and chemical gradients (e.g., in temperature and porewater composition) between various components of the repository and between the repository and geosphere (Gierszewski et al. 2004, Sections 6.3 and 9.2). Afterwards, the gradients diminish until the time period from 10,000 to 100,000 years when quasi-equilibrium conditions are reached and external events, rather than gradients, become the driving force for perturbations to the system. While the containers are subject to some early corrosion, that corrosion has largely stopped sometime between 1,000 and 10,000 years after closure. Pitting and general corrosion are halted due to the lack of oxygen, and stress corrosion cracking is prevented by the lack of tensile stresses in the shell and the high chloride levels in the groundwater. The containers continue to be an effective barrier throughout the 1,000,000-year timeframe, the result being no release of radioactive material from the repository.

For the Defective Container Scenario, investigators ran a Reference Case using best-estimate values for most parameters. The results suggest that estimated dose rates occur far in the future and are well below the ICRP 81 dose constraint and the natural background radiation. The peak calculated dose rate is 2×10^{-4} mSv/yr (0.02 mrem/yr). To analyze the effects of uncertainty, investigators ran 40,368 simulations, the result being calculated dose rates that were still below the ICRP 81 dose constraint and the natural background radiation. Similar results were obtained for sensitivity studies in which specific engineered or natural barriers were assumed to be much less effective. Even in the worst case in which all containers were assumed to fail, the calculated dose rate, though of the same magnitude, did not exceed the ICRP 81 constraint.

The Human Intrusion Scenario leads to potentially significant doses to certain critical groups. However, the deep geologic disposal concept makes inadvertent human intrusion very unlikely and the corresponding probability-weighted annual dose is very low.

Based on the Third Case Study results, the analyzed repository design and hypothetical site appear to provide a good margin of safety for the disposal of high-activity waste in granite. While the containers are the primary barrier to radionuclide release, the other engineered and natural barriers are effective in preventing or delaying the release of most radionuclides. The safety assessment results are consistent with those from other studies in granite such as those from Sweden (Section C-3.3.2) and Finland (Section C-3.3.3).

Fourth Case Study—This safety assessment was conducted to assess key aspects of an updated repository design (Kremer et al. 2011). Compared to the Third Case Study, it considers a different hypothetical site with a shallower repository and revised designs for the underground facilities, floor emplacement, and spent fuel containers.

The geosphere for the Fourth Case Study is again a fractured granitic rock with intermediate permeability. The granite was assumed to extend from the surface to several hundred meters below the 500-m repository depth. The hydrogeology was envisioned as having two distinct zones. The first is a shallow groundwater zone in which advective flow near the surface is driven by topographic gradients. The second is a deep groundwater zone dominated by diffusive flow. In this zone, there are fewer fractures and those fractures are less likely to be interconnected. The groundwater is old, slow moving, and chemically distinct. A discrete fracture network was generated to represent the variations in the fracture patterns down to a depth of 1,500 m.

Unlike the horizontal emplacement used in the previous study, the Fourth Case Study considered a repository layout in which spent fuel containers were placed into vertical boreholes within a series of emplacement rooms. The copper and steel containers, each with a 360-bundle capacity (about 6.9 Mg U per container), are designed to withstand mechanical stresses while providing a corrosion-resistant barrier. The buffer material surrounding the containers consists of compacted bentonite clay. Before closure, all void areas will be backfilled and sealed to isolate the repository from the biosphere.

The safety assessment considered a normal evolution scenario that was a reasonable extrapolation of the present-day site features and receptor lifestyles. Disruptive events such as abnormal or unlikely failures of the containment and isolation systems or inadvertent human intrusion were considered in separate sensitivity studies.

The possibility of having a small number of containers emplaced with undetected penetrating defects, while small, could not be ruled out. As a result, a reference case was defined in which the repository was assumed to meet design specifications except for the presence of two containers having undetected penetrating defects. Groundwater was assumed conservatively to contact the defective containers at 100 years. The resultant peak dose rate to the critical group was 9.7×10^{-5} mSv/yr (9.7×10^{-3} mrem/yr) at more than 1,000,000 years after disposal. This level is well below the ICRP 81 dose rate constraint (0.3 mSv/yr or 30 mrem/yr) as well as the average Canadian natural background radiation (1.7 mSv/yr). The long elapsed time before the peak dose is due to the retention and delay characteristics of the engineered sealing materials and the natural barrier.

Sensitivity studies in which other aspects of engineered or natural barriers were compromised resulted in higher peak dose rates than that of the reference case, but still below the ICRP 81 constraint and the natural background radiation. This outcome was true even for the worst case—all containers fail after 100,000 years—with a peak dose rate of 0.094 mSv/yr (0.94 mrem/yr). In addition, investigators ran 120,000 simulations to evaluate the effect of uncertainty in input parameters. The 99th percentile dose rate was over two orders of magnitude lower than the ICRP 81 constraint. The primary contributor to the average total dose rate was ^{129}I. Since it bypassed the engineered and natural barriers, the human intrusion scenario resulted in potentially significant doses to the few individuals in the critical groups, especially in if the intrusion occurred relatively soon after the assumed loss of institutional control at 300 years. However, the construction of a deep repository with no potable groundwater at depth nor any known mineral or other economic potential would help to minimize the potential for inadvertent human intrusion.

The overall results of Fourth Case Study demonstrate that the hypothetical site and repository design analyzed provide multiple barriers capable of preventing or delaying radionuclide release and transport.

Conclusions—As shown above in the Third and Fourth Case Studies, robust containers are important barriers effectively isolating waste to the repository and preventing or delaying radionuclide transport to the biosphere. However, even in sensitivity cases in which all the containers were assumed to failed, the calculated dose rates to the critical group stayed below the ICRP 81 dose rate constraint and the natural background radiation. This result illustrates the benefits of a multiple barrier system in which the engineered sealing materials and the geosphere are capable of retaining radionuclides or delaying their transport. The safety assessments increase confidence that a deep geologic repository capable of permanently disposing of high-activity waste could be built within crystalline rock in the Canadian Shield.

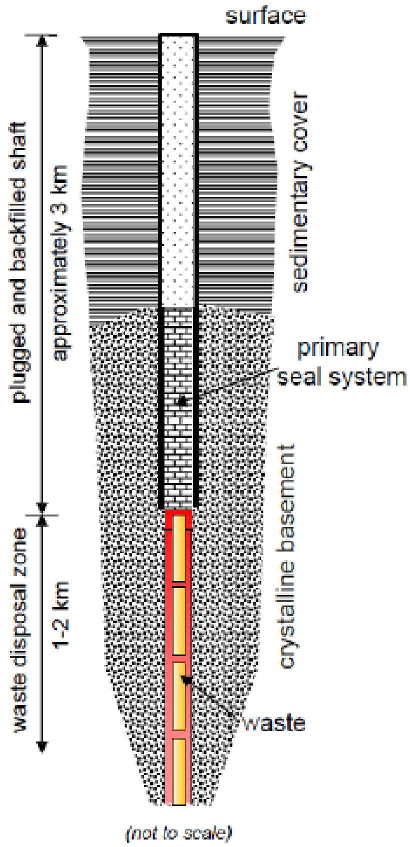
C-3.4 Deep Borehole

C-3.4.1 Generic Deep Borehole Study (United States)

The concept of using deep boreholes for the permanent disposal of nuclear waste has been investigated off and on since the 1950s (Swift et al. 2011). While early proposals emphasized the direct disposal of liquid high-level waste from reprocessing, later analyses have considered solid wastes such as vitrified high-level waste, used nuclear fuel, and surplus weapons-grade plutonium. Even though the general conclusion from the published analyses is that the deep borehole concept has good potential, the United States and other countries around the world have pursued deep, mined geologic repositories. In the past, concerns were raised about the availability of suitable deep drilling technology and about retrievability, whereas mined disposal offered proven mining technology and a more feasible method of waste retrieval. However, the last several decades have seen considerable advances in drilling technology, the result being that the technical challenge of drilling deep boreholes should not be viewed as any greater than that of constructing a deep mine. Retrieval of waste packages prior to sealing of the hole should be possible using similar technology to that used to emplace the packages. After sealing, some portion of the waste could be recovered by drilling operations, but long-term retrievability will likely be more difficult than in mined repositories.

Recent work by Sandia National Laboratories re-examines the deep borehole option with a focus on technical, safety, and performance factors (Brady et al. 2009; Swift et al. 2011). For the purposes of the analyses, the primary performance metric of interest was the mean annual dose to a hypothetical individual. The analyses did not consider the possible consequences of future human intrusion into a deep borehole repository.

The Sandia evaluation is the first quantitative analysis of the deep borehole disposal option using the same safety assessment methodology used for mined geologic repositories. Figure C-10 illustrates the deep borehole disposal concept considered in the analysis. A single borehole is drilled and cased to a depth of 5 km in crystalline basement rock. The waste is emplaced in the lowest 2 km, and a robust sealing system of at least 1-km-thick is installed above the uppermost waste packages. For the safety assessment analyses, it was assumed that there is a shallow aquifer and an associated withdrawal well.



Source: Swift et al. 2011, Figure 1.

Figure C-10. Schematic Illustration of Deep Borehole Disposal of Used Nuclear Fuel or High-Level Radioactive Waste

The borehole concept would allow for the emplacement of about 400 waste canisters assuming a canister length of about 5 m and a borehole disposal interval of 2,000 m. A single disposal site could have multiple boreholes, with thermal considerations dictating the spacing between boreholes. Current technology supports the drilling of 4-km-deep boreholes with a bottom-hole diameter of about 0.5 m. Extending the technology to a total depth of 5 km appears reasonable. Maintaining a bottom-hole diameter of about 0.5 m has the advantage of facilitating direct disposal of intact used fuel assemblies. The following example provides context as to the scale of the number of boreholes needed. Approximately 226,000 fuel assemblies are contained within 65,200 metric tons of commercial used nuclear fuel (Table 3-1). Assuming the emplacement of about 400 assemblies per borehole, the waste could be disposed of in about 565 deep boreholes.

A preliminary analysis of potentially relevant FEPs identified three potential pathways for radionuclide movement given the presence of a sustained upward hydrologic gradient. These potential pathways became the basis for the development of three scenarios:

- **Scenario 1: Transport in the Borehole—**This scenario assumes that higher-than-anticipated permeability in the borehole seals allows radionuclide transport directly up the borehole.
- **Scenario 2: Transport in Disturbed Rock around the Borehole—**This scenario assumes that a high-permeability annulus of fractured rock surrounding the borehole seals allows radionuclide transport upward through the disturbed rock. Eventually the radionuclides migrate to a shallow aquifer from which they are pumped to the biosphere.
- **Scenario 3: Transport in the Surrounding Rock away from the Borehole—**This scenario assumes there is a sufficiently high permeability in the fracture zones, faults, or a combination of the two in the crystalline basement rock and overlying sedimentary units to allow radionuclide transport upward through the rocks to a shallow aquifer and then to the biosphere. This pathway requires some degree of connectivity between the basement rock and the overlying sedimentary units.

A preliminary safety assessment was conducted combining the first two scenarios by considering the borehole seal and the annulus of fractured rock around the borehole to be one, cylindrical element. The cylindrical element was assigned an effective permeability based on the permeabilities of the seal and the fractured rock. The safety assessment did not address the third scenario. It was assumed that such high-permeability features could be detected by downhole testing, and the borehole could therefore be plugged and abandoned before any waste was emplaced.

The results for the combined scenario treatment indicate that most radionuclides will not leave the borehole or borehole's immediate vicinity. The fuel and the majority of the radionuclides in it will be thermodynamically stable, and as such, will resist dissolution into borehole fluids. Sorption on the rock and/or borehole fill material will retard transport for many radionuclides. Further, the period of time in which there is advective transport of radionuclides up the borehole is only about 200 years, the result of thermally driven flow. In the subsequent ambient conditions, there is no upward advective transport, only diffusive transport. Over the 1,000,000 year period, diffusive transport cannot move radionuclides through the various media more than about 200 m. On this timescale, the vast majority of radionuclides will have decayed.

The results show that ^{129}I and ^{36}Cl, the only two radionuclides assumed to have no retardation, are the only radionuclides with non-zero concentrations 1,000 m above the waste disposal zone in the sealed borehole. At that location, the low ^{129}I and ^{36}Cl concentrations (1.0×10^{-7} mg/L and 1.9×10^{-10} mg/L, respectively) represent the leading edge of the dispersive transport front. The travel time for radionuclides to actually reach the withdrawal well from the top of the sealed borehole zone is about 8,000 years. Accounting for that travel time, the peak dose to the reasonably maximally exposed individual is 3.4×10^{-10} mrem/yr, occurring at 8,200 years. The primary contributor is ^{129}I, with ^{36}Cl providing a minor contribution.

After years of intermittent consideration, the deep borehole disposal of high-activity waste is being re-examined as a viable option due in part because of advances in drilling technology as well as possible implementation and cost advantages (i.e., there is widespread geographic potential for deep boreholes, and there can be numerous deep borehole disposal facility locations). The results of the preliminary safety assessment conducted by Sandia National Laboratories (Brady et al. 2009; Swift et al. 2011) have shown that the deep borehole disposal option has the potential for excellent long-term performance in terms of permanent waste isolation.

Appendix D

FEP Analysis Results

Appendix D—FEP Analysis Results

D-1. Introduction

This Appendix presents the results of preliminary FEP identification supporting the generic safety case, as described in Section 4.2.2. The results are provided in two tables: Table D-1 shows the organizational structure and hierarchical numbering scheme for categorizing the FEPs; and Table D-2 lists the comprehensive list of 208 FEPs for generic geologic disposal, applicable to each disposal option. The FEPs are consistent with the list of 208 FEPs developed by the Used Fuel Disposition (UFD) Campaign (Freeze et al. 2010; Freeze et al. 2011). The UFD FEP numbering hierarchy follows the classification and numbering scheme developed for the International FEP Database (Organisation for Economic Co-operation and Development 1999a). The UFD FEP numbers have the form x.x.xx.xx, where the first three levels (x.x.xx) correspond to a physical location or set of similar processes within the disposal system (Figure 4-1). The fourth level of the UFD FEP number (x.x.xx.xx) represents subjective, sequential numbering for multiple FEPs mapped to one of the third levels.

Table D-1. UFD FEP Classification and Numbering Hierarchy

0.0.00.00	Assessment Basis				
1.0.00.00	External Factors				
		1.1.00.00	Repository Issues		
		1.2.00.00	Geological Processes and Effects		
				1.2.01.00	Long-term Processes
				1.2.03.00	Seismic Activity
				1.2.04.00	Igneous Activity
		1.3.00.00	Climatic Processes and Effects		
		1.4.00.00	Future Human Actions		
		1.5.00.00	Other		
2.0.00.00	Disposal System Factors				
		2.1.00.00	Wastes and Engineered Features		
				2.1.01.00	Inventory
				2.1.02.00	Waste Form
				2.1.03.00	Waste Container
				2.1.04.00	Buffer / Backfill
				2.1.05.00	Seals
				2.1.06.00	Other EBS Materials
				2.1.07.00	Mechanical Processes
				2.1.08.00	Hydrologic Processes
				2.1.09.00	Chemical Processes – Chemistry
				2.1.09.50	Chemical Processes – Transport
				2.1.10.00	Biological Processes
				2.1.11.00	Thermal Processes
				2.1.12.00	Gas Sources and Effects
				2.1.13.00	Radiation Effects
				2.1.14.00	Nuclear Criticality

Table D-1. UFD FEP Classification and Numbering Hierarchy (continued)

	2.2.00.00	Geological Environment	
		2.2.01.00	Excavation Disturbed Zone (EDZ)
		2.2.02.00	Host Rock
		2.2.03.00	Other Geologic Units
		2.2.05.00	Flow and Transport Pathways
		2.2.07.00	Mechanical Processes
		2.2.08.00	Hydrologic Processes
		2.2.09.00	Chemical Processes – Chemistry
		2.2.09.50	Chemical Processes – Transport
		2.2.10.00	Biological Processes
		2.2.11.00	Thermal Processes
		2.2.12.00	Gas Sources and Effects
		2.2.14.00	Nuclear Criticality
	2.3.00.00	Surface Environment	
		2.3.01.00	Surface Characteristics
		2.3.07.00	Mechanical Processes
		2.3.08.00	Hydrologic Processes
		2.3.09.00	Chemical Processes – Chemistry
		2.3.09.50	Chemical Processes – Transport
		2.3.10.00	Biological Processes
		2.3.11.00	Thermal Processes
	2.4.00.00	Human Behavior	
		2.4.01.00	Human Characteristics
		2.4.04.00	Lifestyle
		2.4.08.00	Land and Water Use
3.0.00.00	Radionuclide / Contaminant Factors (Biosphere)		
	3.1.00.00	Contaminant Characteristics	
	3.2.00.00	Release / Migration Factors	
	3.3.00.00	Exposure Factors	
		3.3.01.00	Radionuclide / Contaminant Concentrations
		3.3.04.00	Exposure Modes
		3.3.06.00	Toxicity Effects

In Table D-2, each UFD FEP is defined by a "Description" at a broad level of detail such that it is potentially applicable to all of the disposal options. Each UFD FEP is further defined by additional details under "Associated Processes". The level of detail captured by the FEP Descriptions and Associated Processes is appropriate for supporting a generic safety case. Mapping to the FEPs from Sandia National Laboratories (SNL) (2008b) is also shown.

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
0.0.00.00	0. ASSESSMENT BASIS		
0.1.02.01	Timescales of Concern		0.1.02.00.0A
0.1.03.01	Spatial Domain of Concern		0.1.03.00.0A
0.1.09.01	Regulatory Requirements and Exclusions		0.1.09.00.0A
0.1.10.01	Model Issues	- Conceptual model - Mathematical implementation - Geometry and dimensionality - Process coupling - Boundary and initial conditions	0.1.10.00.0A
0.1.10.02	Data Issues	- Parameterization and values - Correlations - Uncertainty	0.1.10.00.0A
1.0.00.00	1. EXTERNAL FACTORS		
1.1.00.00	1. REPOSITORY ISSUES		
1.1.01.01	Open Boreholes	- Site investigation boreholes (open, improperly sealed) - Preclosure and postclosure monitoring boreholes - Enhanced flow pathways from EBS	1.1.01.01.0A 1.1.11.00.0A
1.1.02.01	Chemical Effects from Preclosure Operations - In EBS - In EDZ - In Host Rock	- Water contaminants (explosives residue, diesel, organics, etc.) - Water chemistry different than host rock (e.g., oxiding) - Undesirable materials left - Accidents and unplanned events	1.1.02.00.0A 1.1.02.03.0A 1.1.12.01.0A 2.2.01.01.0B
1.1.02.02	Mechanical Effects from Preclosure Operations - In EBS - In EDZ - In Host Rock	- Creation of excavation-disturbed zone (EDZ) - Stress relief - Boring and blasting effects - Rock reinforcement effects (drillholes) - Accidents and unplanned events - Enhanced flow pathways [see also Evolution of EDZ in 2.2.01.01]	1.1.01.01.0B 1.1.02.00.0B 1.1.12.01.0A 2.2.01.01.0A
1.1.02.03	Thermal-Hydrologic Effects from Preclosure Operations - In EBS - In EDZ - In Host Rock	- Site flooding - Preclosure ventilation - Accidents and unplanned events	1.1.02.01.0A 1.1.02.02.0A 1.1.12.01.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
1.1.08.01	Deviations from Design and Inadequate Quality Control	- Error in waste emplacement (waste forms, waste packages, waste package support materials) - Error in EBS component emplacement (backfill, seals, liner) - Inadequate excavation / construction (planning, schedule, implementation) - Aborted / incomplete closure of repository - Material and/or component defects	1.1.03.01.0A 1.1.03.01.0B 1.1.04.01.0A 1.1.07.00.0A 1.1.08.00.0A 1.1.09.00.0A
1.1.10.01	Control of Repository Site	- Active controls (controlled area) - Retention of records - Passive controls (markers)	1.1.05.00.0A 1.1.10.00.0A
1.1.13.01	Retrievability		1.1.13.00.0A
1.2.00.00	2. GEOLOGICAL PROCESSES AND EFFECTS		
1.2.01.00	2.01. LONG-TERM PROCESSES		
1.2.01.01	Tectonic Activity – Large Scale	- Uplift - Folding	1.2.01.01.0A
1.2.01.02	Subsidence		2.2.06.04.0A
1.2.01.03	Metamorphism	- Structural changes due to natural heating and/or pressure	1.2.05.00.0A
1.2.01.04	Diagenesis	- Mineral alteration due to natural processes	1.2.08.00.0A
1.2.01.05	Diapirism	- Plastic flow of rocks under lithostatic loading - Salt / evaporates - Clay	1.2.09.00.0A 1.2.09.01.0A
1.2.01.06	Large-Scale Dissolution		1.2.09.02.0A
1.2.03.00	2.03.SEISMIC ACTIVITY		
1.2.03.01	Seismic Activity Impacts EBS and/or EBS Components	- Mechanical damage to EBS (from ground motion, rockfall, drift collapse, fault displacement) [see also Mechanical Impacts in 2.1.07.04, 2.1.07.05, 2.1.07.06, 2.1.07.07, 2.1.07.08, and 2.1.07.10]	1.2.02.03.0A 1.2.03.02.0A 1.2.03.02.0B 1.2.03.02.0C 1.2.03.03.0A
1.2.03.02	Seismic Activity Impacts Geosphere - Host Rock - Other Geologic Units	- Altered flow pathways and properties - Altered stress regimes (faults, fractures) [see also Alterations and Impacts in 2.2.05.01, 2.2.05.02, 2.2.05.03, 2.1.07.01, and 2.1.07.02]	1.2.03.03.0A 1.2.10.01.0A 2.2.06.01.0A 2.2.06.02.0A 2.2.06.02.0B 2.2.06.03.0A
1.2.03.03	Seismic Activity Impacts Biosphere - Surface Environment - Human Behavior	- Altered surface characteristics - Altered surface transport pathways - Altered recharge	2.3.01.00.0A 2.3.11.03.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
1.2.04.00	2.04. IGNEOUS ACTIVITY		
1.2.04.01	Igneous Activity Impacts EBS and/or EBS Components	- Mechanical damage to EBS (from igneous intrusion) - Chemical interaction with magmatic volatiles - Transport of radionuclides (in magma, pyroclasts, vents) [see also Mechanical Impacts in 2.1.07.04, 2.1.07.05, 2.1.07.06, 2.1.07.07, and 2.1.07.08]	1.2.04.03.0A 1.2.04.04.0A 1.2.04.04.0B 1.2.04.05.0A 1.2.04.06.0A
1.2.04.02	Igneous Activity Impacts Geosphere - Host Rock - Other Geologic Units	- Altered flow pathways and properties - Altered stress regimes (faults, fractures) - Igneous intrusions - Altered thermal and chemical conditions [see also Alterations and Impacts in 2.2.05.01, 2.2.05.02, 2.2.05.03, 2.1.07.01, 2.1.07.02, 2.2.09.03, 2.2.11.06 and 2.2.11.07]	1.2.04.02.0A 1.2.10.02.0A
1.2.04.03	Igneous Activity Impacts Biosphere - Surface Environment - Human Behavior	- Altered surface characteristics - Altered surface transport pathways - Altered recharge - Ashfall and ash redistribution	1.2.04.07.0A 1.2.04.07.0B 1.2.04.07.0C 2.3.01.00.0A 2.3.11.03.0A
1.3.00.00	3. CLIMATIC PROCESSES AND EFFECTS		
1.3.01.01	Climate Change - Natural - Anthropogenic	- Variations in precipitation and temperature - Long-term global (sea level, …) - Short-term regional and local - Seasonal local (flooding, storms, …) [see also Human Influences on Climate in 1.4.01.01] [contributes to Precipitation in 2.3.08.01, Surface Runoff and Evapotranspiration in 2.3.08.02]	1.3.01.00.0A
1.3.04.01	Periglacial Effects	- Permafrost - Seasonal freeze/thaw	1.3.04.00.0A
1.3.05.01	Glacial and Ice Sheet Effects	- Glaciation - Isostatic depression - Melt water	1.3.05.00.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
1.4.00.00	4. FUTURE HUMAN ACTIONS		
1.4.01.01	Human Influences on Climate - Intentional - Accidental	- Variations in precipitation and temperature - Global, regional, and/or local - Greenhouse gases, ozone layer failure [contributes to Climate Change in 1.3.01.01]	1.4.01.00.0A 1.4.01.01.0A 1.4.01.02.0A 1.4.01.04.0A
1.4.02.01	Human Intrusion - Deliberate - Inadvertent	- Drilling (resource exploration, …) - Mining / tunneling - Unintrusive site investigation (airborne, surface-based, …) [see also Control of Repository Site in 1.1.10.01]	1.4.02.01.0A 1.4.02.02.0A 1.4.03.00.0A 1.4.04.00.0A 1.4.04.01.0A 1.4.05.00.0A 3.3.06.01.0A
1.4.11.01	Explosions and Crashes from Human Activities	- War - Sabotage - Testing - Resource exploration / exploitation - Aircraft	1.4.11.00.0A
1.5.00.00	5. OTHER		
1.5.01.01	Meteorite Impact	- Cratering, host rock removal - Exhumation of waste - Alteration of flow pathways	1.5.01.01.0A
1.5.01.02	Extraterrestrial Events	- Solar systems (supernova) - Celestial activity (sun - solar flares, gamma-ray bursters; moon – earth tides) - Alien life forms	1.5.01.02.0A 1.5.03.02.0A
1.5.03.01	Earth Planetary Changes	- Changes in earth's magnetic field - Changes in earth's gravitational field (tides) - Changes in ocean currents	1.5.03.01.0A 1.5.03.02.0A
2.0.00.00	2. DISPOSAL SYSTEM FACTORS		
2.1.00.00	1. WASTES AND ENGINEERED FEATURES		
2.1.01.00	1.01. INVENTORY		
2.1.01.01	Waste Inventory - Radionuclides - Non-Radionuclides	- Composition - Enrichment / Burn-up	2.1.01.01.0A
2.1.01.02	Radioactive Decay and Ingrowth	- Decay chains - Decay products - Neutron activation	3.1.01.01.0A
2.1.01.03	Heterogeneity of Waste Inventory - Waste Package Scale - Repository Scale	- Composition - Enrichment / Burn-up - Damaged Area	2.1.01.03.0A 2.1.01.04.0A
2.1.01.04	Interactions Between Co-Located Waste		2.1.01.02.0A 2.1.01.02.0B

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.02.00	1.02. WASTE FORM		
2.1.02.01	SNF (Commercial, DOE) Degradation - Alteration / Phase Separation - Dissolution / Leaching - Radionuclide Release	Degradation is dependent on: - Composition - Geometry / Structure - Enrichment / Burn-up - Surface Area - Gap and Grain Fraction - Damaged Area - THC Conditions [see also Mechanical Impact in 2.1.07.06 and Thermal-Mechanical Effects in 2.1.11.06]	2.1.02.02.0A 2.1.02.01.0A 2.1.02.28.0A 2.1.02.07.0A
2.1.02.02	HLW (Glass, Ceramic, Metal) Degradation - Alteration / Phase Separation - Dissolution / Leaching - Radionuclide Release	Degradation is dependent on: - Composition - Geometry / Structure - Surface Area - Damaged / Cracked Area - Mechanical Impact - THC Conditions [see also Mechanical Impact in 2.1.07.07 and Thermal-Mechanical Effects in 2.1.11.06]	2.1.02.03.0A 2.1.02.05.0A
2.1.02.03	Degradation of Organic/Cellulosic Materials in Waste	[see also Complexation in EBS in 2.1.09.54]	2.1.02.10.0A
2.1.02.04	HLW (Glass, Ceramic, Metal) Recrystallization		2.1.02.06.0A
2.1.02.05	Pyrophoricity or Flammable Gas from SNF or HLW	[see also Gas Explosions in EBS in 2.1.12.04]	2.1.02.08.0A 2.1.02.29.0A
2.1.02.06	SNF Cladding Degradation and Failure	- Initial damage - General Corrosion - Microbially Influenced Corrosion - Localized Corrosion - Enhanced Corrosion (silica, fluoride) - Stress Corrosion Cracking - Hydride Cracking - Unzipping - Creep - Internal Pressure - Mechanical Impact	2.1.02.11.0A 2.1.02.12.0A 2.1.02.13.0A 2.1.02.14.0A 2.1.02.15.0A 2.1.02.16.0A 2.1.02.17.0A 2.1.02.18.0A 2.1.02.27.0A 2.1.02.21.0A 2.1.02.22.0A 2.1.02.23.0A 2.1.02.25.0A 2.1.02.25.0B 2.1.02.19.0A 2.1.02.26.0A 2.1.02.20.0A 2.1.02.24.0A 2.1.09.03.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.03.00	1.03. WASTE CONTAINER		
2.1.03.01	Early Failure of Waste Packages	- Manufacturing defects - Improper sealing [see also Deviations from Design in 1.1.08.01]	2.1.03.08.0A
2.1.03.02	General Corrosion of Waste Packages	- Dry-air oxidation - Humid-air corrosion - Aqueous phase corrosion - Passive film formation and stability	2.1.03.01.0A
2.1.03.03	Stress Corrosion Cracking (SCC) of Waste Packages	- Crack initiation, growth and propagation - Stress distribution around cracks	2.1.03.02.0A
2.1.03.04	Localized Corrosion of Waste Packages	- Pitting - Crevice corrosion - Salt deliquescence [see also 2.1.09.06 Chemical Interaction with Backfill]	2.1.03.03.0A 2.1.09.28.0A
2.1.03.05	Hydride Cracking of Waste Packages	- Hydrogen diffusion through metal matrix - Crack initiation and growth in metal hydride phases	2.1.03.04.0A
2.1.03.06	Microbially Influenced Corrosion (MIC) of Waste Packages		2.1.03.05.0A
2.1.03.07	Internal Corrosion of Waste Packages Prior to Breach		2.1.03.06.0A
2.1.03.08	Evolution of Flow Pathways in Waste Packages	- Evolution of physical form of waste package - Plugging of cracks in waste packages [see also Evolution of Flow Pathways in EBS in 2.1.08.06, Mechanical Impacts in 2.1.07.05, 2.1.07.06, and 2.1.07.07, Thermal-Mechanical Effects in 2.1.11.06 and 2.1.11.07]	2.1.03.10.0A 2.1.03.11.0A
2.1.04.00	1.04. BUFFER / BACKFILL		
2.1.04.01	Evolution and Degradation of Backfill	- Alteration - Thermal expansion / Degradation - Swelling / Compaction - Erosion / Dissolution - Evolution of backfill flow pathways [see also Evolution of Flow Pathways in EBS in 2.1.08.06, Mechanical Impact in 2.1.07.04, Thermal-Mechanical Effects in 2.1.11.08, Chemical Interaction in 2.1.09.06]	2.1.04.05.0A 2.1.04.03.0A
2.1.05.00	1.05. SEALS		
2.1.05.01	Degradation of Seals	- Alteration / Degradation / Cracking - Erosion / Dissolution [see also Mechanical Impact in 2.1.07.08, Thermal-Mechanical Effects in 2.1.11.09, Chemical Interaction in 2.1.09.08]	2.1.05.03.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.06.00	1.06. OTHER EBS MATERIALS		
2.1.06.01	Degradation of Liner / Rock Reinforcement Materials in EBS	- Alteration / Degradation / Cracking - Corrosion - Erosion / Dissolution / Spalling [see also Mechanical Impact in 2.1.07.08, Thermal-Mechanical Effects in 2.1.11.09, Chemical Interaction in 2.1.09.07]	2.1.06.02.0A
2.1.07.00	1.07. MECHANICAL PROCESSES		
2.1.07.01	Rockfall	- Dynamic loading (block size and velocity) [see also Mechanical Effects on Host Rock in 2.2.07.01]	2.1.07.01.0A
2.1.07.02	Drift Collapse	- Static loading (rubble volume) - Alteration of seepage - Alteration of EBS flow pathways - Alteration of EBS thermal environment [see also Evolution of Flow Pathways in EBS in 2.1.08.06, Chemical Effects of Drift Collapse in 2.1.09.12, and Effects of Drift Collapse on TH in 2.1.11.04, Mechanical Effects on Host Rock in 2.2.07.01]	2.1.07.02.0A 1.2.03.02.0D
2.1.07.03	Mechanical Effects of Backfill	- Protection of other EBS components from rockfall / drift collapse	2.1.04.04.0A
2.1.07.04	Mechanical Impact on Backfill	- Rockfall / Drift collapse - Hydrostatic pressure - Internal gas pressure [see also Degradation of Backfill in 2.1.04.01 and Thermal-Mechanical Effects in 2.1.11.08]	2.1.04.05.0A
2.1.07.05	Mechanical Impact on Waste Packages	- Rockfall / Drift collapse - Waste package movement - Hydrostatic pressure - Internal gas pressure - Swelling corrosion products [see also Thermal-Mechanical Effects in 2.1.11.07]	2.1.03.07.0A 2.1.07.04.0A 2.1.09.03.0B
2.1.07.06	Mechanical Impact on SNF Waste Form	- Drift collapse - Swelling corrosion products [see also Thermal-Mechanical Effects in 2.1.11.06]	2.1.07.02.0A 2.1.09.03.0B
2.1.07.07	Mechanical Impact on HLW Waste Form	- Drift collapse - Swelling corrosion products [see also Thermal-Mechanical Effects in 2.1.11.06]	2.1.07.02.0A 2.1.09.03.0B

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.07.08	Mechanical Impact on Other EBS Components - Seals - Liner / Rock Reinforcement Materials - Waste Package Support Materials	- Rockfall / Drift collapse - Movement - Hydrostatic pressure - Swelling corrosion products [see also Thermal-Mechanical Effects in 2.1.11.09]	2.1.07.02.0A 2.1.09.03.0C
2.1.07.09	Mechanical Effects at EBS Component Interfaces	- Component-to-component contact (static or dynamic)	2.1.06.07.0B 2.1.08.15.0A
2.1.07.10	Mechanical Degradation of EBS	- Floor buckling - Fault displacement - Initial damage from excavation / construction - Consolidation of EBS components - Degradation of waste package support structure - Alteration of EBS flow pathways [see also Mechanical Effects from Preclosure in 1.1.02.02, Evolution of Flow Pathways in EBS in 2.1.08.06, Drift Collapse in 2.1.07.02, Degradation in 2.1.04.01, 2.1.05.01, and 2.1.06.01, and Mechanical Effects on Host Rock in 2.2.07.01]	2.1.06.05.0B 2.1.07.06.0A 1.2.02.03.0A 2.1.08.15.0A
2.1.08.00	1.08. HYDROLOGIC PROCESSES		
2.1.08.01	Flow Through the EBS	- Saturated / Unsaturated flow - Preferential flow pathways - Density effects on flow - Initial hydrologic conditions - Flow pathways out of EBS [see also Open Boreholes in 1.1.01.01, Thermal-Hydrologic Effects from Preclosure in 1.1.02.03, Flow in Waste Packages in 2.1.08.02, Flow in Backfill in 2.1.08.03, Flow through Seals 2.1.08.04, Flow through Liner in 2.1.08.05, Thermal Effects on Flow in 2.1.11.10, Effects of Gas on Flow in 2.1.12.02]	2.1.08.09.0A 2.1.08.07.0A 2.1.08.05.0A
2.1.08.02	Flow In and Through Waste Packages	- Saturated / Unsaturated flow - Movement as thin films or droplets	2.1.03.10.0A 2.1.03.11.0A
2.1.08.03	Flow in Backfill	- Fracture / Matrix flow	2.1.04.01.0A
2.1.08.04	Flow Through Seals	- Fracture / Matrix flow	2.1.05.01.0A
2.1.08.05	Flow Through Liner / Rock Reinforcement Materials in EBS		2.1.06.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.08.06	Alteration and Evolution of EBS Flow Pathways	- Drift collapse - Degradation/consolidation of EBS components - Plugging of flow pathways - Formation of corrosion products - Water ponding [see also Evolution of Flow Pathways in WPs in 2.1.03.08, Evolution of Backfill in 2.1.04.01, Drift Collapse in 2.1.07.02, and Mechanical Degradation of EBS in 2.1.07.10]	2.1.08.12.0A 2.1.08.15.0A 2.1.03.10.0A 2.1.03.11.0A 2.1.09.02.0A
2.1.08.07	Condensation Forms in Repository - On Tunnel Roof / Walls - On EBS Components	- Heat transfer (spatial and temporal distribution of temperature and relative humidity) - Dripping - Moisture movement [see also Heat Generation in EBS in 2.1.11.01, Effects on EBS Thermal Environment in 2.1.11.03 and 2.1.11.04]	2.1.08.04.0A 2.1.08.04.0B
2.1.08.08	Capillary Effects in EBS	- Wicking - Capillary barrier - Osmotic binding	2.1.08.06.0A
2.1.08.09	Influx/Seepage Into the EBS	- Water influx rate (spatial and temporal distribution) [see also Open Boreholes in 1.1.01.01, Thermal Effects on Flow in EBS in 2.1.11.10, Flow Through Host Rock in 2.2.08.01, Effects of Excavation on Flow in 2.2.08.04]	2.1.08.01.0A
2.1.09.00	1.09. CHEMICAL PROCESSES - CHEMISTRY		
2.1.09.01	Chemistry of Water Flowing into the Repository	- Chemistry of influent water (spatial and temporal distribution) [See also Chemistry in Host Rock 2.2.09.01]	2.2.08.12.0A 2.1.08.01.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.09.02	Chemical Characteristics of Water in Waste Packages	- Water composition (radionuclides, dissolved species, …) - Initial void chemistry (air / gas) - Water chemistry (pH, ionic strength, pCO2, ..) - Reduction-oxidation potential - Reaction kinetics - Influent chemistry (from tunnels and/or backfill) [see also Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04] - Evolution of water chemistry / interaction with waste packages	2.1.09.01.0B 2.1.02.09.0A 2.2.08.12.0B 2.1.09.06.0A 2.1.09.07.0A
2.1.09.03	Chemical Characteristics of Water in Backfill	- Water composition (radionuclides, dissolved species, …) - Water chemistry (pH, ionic strength, pCO2, ..) - Reduction-oxidation potential - Reaction kinetics - Influent chemistry (from tunnels and/or waste package) [see also Chemistry in Waste Packages in 2.1.09.02, Chemistry in Tunnels in 2.1.09.04] - Evolution of water chemistry / interaction with backfill	2.1.04.02.0A 2.1.09.01.0A 2.1.09.06.0B 2.1.09.07.0B
2.1.09.04	Chemical Characteristics of Water in Tunnels	- Water composition (radionuclides, dissolved species, …) - Water chemistry (pH, ionic strength, pCO2, ..) - Reduction-oxidation potential - Reaction kinetics - Influent chemistry (from near-field host rock) - Initial chemistry (from construction / emplacement) [see also Chemical Effects from Preclosure in 1.1.02.01, Chemistry of Water Flowing in 2.1.09.01, Chemistry in Waste Packages in 2.1.09.02, Chemistry in Backfill in 2.1.09.03] - Evolution of water chemistry / interaction with seals, liner/rock reinforcement materials, waste package support materials	2.1.09.01.0A 2.1.09.06.0B 2.1.09.07.0B
2.1.09.05	Chemical Interaction of Water with Corrosion Products - In Waste Packages - In Backfill - In Tunnels	- Corrosion product formation and composition (waste form, waste package internals, waste package) - Evolution of water chemistry in waste packages, in backfill, and in tunnels [contributes to Chemistry in Waste Packages in 2.1.09.02, Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.09.02.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.09.06	Chemical Interaction of Water with Backfill - On Waste Packages - In Backfill - In Tunnels	- Backfill composition and evolution (bentonite, crushed rock, ...) - Evolution of water chemistry in backfill, and in tunnels - Enhanced degradation of waste packages (crevice formation) [contributes to Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04, Localized Corrosion of WPs in 2.1.03.04]	2.1.04.02.0A
2.1.09.07	Chemical Interaction of Water with Liner / Rock Reinforcement and Cementitious Materials in EBS - In Backfill - In Tunnels	- Liner composition and evolution (concrete, metal, ...) - Rock reinforcement material composition and evolution (grout, rock bolts, mesh, ...) - Other cementitious materials composition and evolution - Evolution of water chemistry in backfill, and in tunnels [contributes to Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.06.01.0A
2.1.09.08	Chemical Interaction of Water with Other EBS Components - In Waste Packages - In Tunnels	- Seals composition and evolution - Waste Package Support composition and evolution (concrete, metal, ...) - Other EBS components (other metals (copper), ...) - Evolution of water chemistry in backfill, and in tunnels [contributes to Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.06.05.0D 2.1.03.09.0A
2.1.09.09	Chemical Effects at EBS Component Interfaces	- Component-to-component contact (chemical reactions) - Consolidation of EBS components	2.1.06.07.0A 2.1.08.15.0A
2.1.09.10	Chemical Effects of Waste-Rock Contact	- Waste-to-host rock contact (chemical reactions) - Component-to-host rock contact (chemical reactions)	2.1.09.11.0A 2.2.01.02.0B
2.1.09.11	Electrochemical Effects in EBS	- Enhanced metal corrosion	2.1.09.09.0A 2.1.09.27.0A
2.1.09.12	Chemical Effects of Drift Collapse	- Evolution of water chemistry in backfill and in tunnels (from altered seepage, from altered thermal-hydrology) [contributes to Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	1.2.03.02.0E
2.1.09.13	Radionuclide Speciation and Solubility in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Dissolved concentration limits - Limited dissolution due to inclusion in secondary phase - Enhanced dissolution due to alpha recoil [controlled by Chemistry in Waste Packages in 2.1.09.02, Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.09.04.0A 2.1.09.10.0A 2.1.02.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.09.50	1.09. CHEMICAL PROCESSES - TRANSPORT		
2.1.09.51	Advection of Dissolved Radionuclides in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Flow pathways and velocity - Advective properties (porosity, tortuosity) - Dispersion - Saturation [see also Gas Phase Transport in 2.1.12.03]	2.1.09.08.0B 2.1.04.09.0A 2.1.09.27.0A
2.1.09.52	Diffusion of Dissolved Radionuclides in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Gradients (concentration, chemical potential) - Diffusive properties (diffusion coefficients) - Flow pathways and velocity - Saturation	2.1.09.08.0A 2.1.04.09.0A 2.1.09.27.0A
2.1.09.53	Sorption of Dissolved Radionuclides in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Surface complexation properties - Flow pathways and velocity - Saturation [see also Chemistry in Waste Packages in 2.1.09.02, Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.09.05.0A 2.1.04.09.0A 2.1.09.27.0A
2.1.09.54	Complexation in EBS	- Formation of organic complexants (humates, fulvates, organic waste) - Enhanced transport of radionuclides associated with organic complexants [see also Degradation of Organics in Waste in 2.1.02.03, see Radionuclide Speciation in 2.1.09.13 for inorganic complexation]	2.1.09.13.0A
2.1.09.55	Formation of Colloids in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Formation of intrinsic colloids - Formation of pseudo colloids (host rock fragments, waste form fragments, corrosion products, microbes) - Formation of co-precipitated colloids - Sorption/attachment of radionuclides to colloids (clay, silica, waste form, FeOx, microbes)	2.1.09.15.0A 2.1.09.16.0A 2.1.09.17.0A 2.1.09.18.0A 2.1.09.25.0A
2.1.09.56	Stability of Colloids in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Chemical stability of attachment (dependent on water chemistry) - Mechanical stability of colloid (dependent on colloid size, gravitational settling)	2.1.09.23.0A 2.1.09.26.0A 2.1.09.21.0A
2.1.09.57	Advection of Colloids in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Flow pathways and velocity - Advective properties (porosity, tortuosity) - Dispersion - Saturation - Colloid concentration	2.1.09.19.0B 2.1.04.09.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.09.58	Diffusion of Colloids in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Gradients (concentration, chemical potential) - Diffusive properties (diffusion coefficients) - Flow pathways and velocity - Saturation - Colloid concentration	2.1.09.24.0A 2.1.04.09.0A
2.1.09.59	Sorption of Colloids in EBS - In Waste Form - In Waste Package - In Backfill - In Tunnel	- Surface complexation properties - Flow pathways and velocity - Saturation - Colloid concentration [see also Chemistry in Waste Packages in 2.1.09.02, Chemistry in Backfill in 2.1.09.03, Chemistry in Tunnels in 2.1.09.04]	2.1.09.19.0A 2.1.04.09.0A
2.1.09.60	Sorption of Colloids at Air-Water Interface in EBS		2.1.09.22.0A
2.1.09.61	Filtration of Colloids in EBS	- Physical filtration or trapping (dependent on flow pathways, colloid size) - Electrostatic filtration	2.1.09.20.0A 2.1.09.21.0A
2.1.09.62	Radionuclide Transport Through Liners and Seals	- Advection - Dispersion - Diffusion - Sorption [contributes to Radionuclide release from EBS in 2.1.09.63]	2.1.05.02.0A
2.1.09.63	Radionuclide Release from the EBS - Dissolved - Colloidal - Gas Phase	- Spatial and temporal distribution of releases to the host rock (due to varying flow pathways and velocities, varying component degradation rates, varying transport properties) [contributions from Dissolved in 2.1.09.51/52/53, Colloidal in 2.1.09.57/58/59, Gas Phase in 2.1.12.03, Liners and Seals in 2.1.09.62]	2.2.07.06.0A 2.2.07.06.0B
2.1.10.00	1.10. BIOLOGICAL PROCESSES		
2.1.10.01	Microbial Activity in EBS - Natural - Anthropogenic	- Effects on corrosion - Formation of complexants - Formation of microbial colloids - Formation of biofilms - Biodegradation - Biomass production - Bioaccumulation [see also Microbially Influenced Corrosion in 2.1.03.06, Complexation in EBS in 2.1.09.54, Radiological Mutation of Microbes in 2.1.13.03]	2.1.10.01.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.11.00	1.11. THERMAL PROCESSES		
2.1.11.01	Heat Generation in EBS	- Heat transfer (spatial and temporal distribution of temperature and relative humidity) [see also Thermal-Hydrologic Effects from Preclosure in 1.1.02.03, Waste Inventory in 2.1.01.01]	2.1.11.01.0A 2.1.11.02.0A
2.1.11.02	Exothermic Reactions in EBS	- Oxidation of SNF - Hydration of concrete	2.1.11.03.0A
2.1.11.03	Effects of Backfill on EBS Thermal Environment	- Thermal blanket - Condensation	2.1.04.04.0A
2.1.11.04	Effects of Drift Collapse on EBS Thermal Environment	- Thermal blanket - Condensation	1.2.03.02.0D
2.1.11.05	Effects of Influx (Seepage) on Thermal Environment	- Temperature and relative humidity (spatial and temporal distribution) [see also Influx/Seepage into EBS in 2.1.08.09]	2.1.08.01.0B 2.1.08.01.0A
2.1.11.06	Thermal-Mechanical Effects on Waste Form and In-Package EBS Components	- Alteration - Cracking - Thermal expansion / stress	2.1.11.05.0A
2.1.11.07	Thermal-Mechanical Effects on Waste Packages	- Thermal sensitization / phase changes - Cracking - Thermal expansion / stress / creep	2.1.07.05.0A 2.1.11.06.0A 2.1.11.07.0A
2.1.11.08	Thermal-Mechanical Effects on Backfill	- Alteration - Cracking - Thermal expansion / stress	2.1.11.07.0A 2.1.04.04.0A
2.1.11.09	Thermal-Mechanical Effects on Other EBS Components - Seals - Liner / Rock Reinforcement Materials - Waste Package Support Structure	- Alteration - Cracking - Thermal expansion / stress	2.1.11.07.0A
2.1.11.10	Thermal Effects on Flow in EBS	- Altered influx/seepage - Altered saturation / relative humidity (dry-out, resaturation) - Condensation	2.1.08.03.0A 2.1.08.11.0A 2.1.11.09.0A
2.1.11.11	Thermally-Driven Flow (Convection) in EBS	- Convection	2.1.11.09.0B 2.1.11.09.0C
2.1.11.12	Thermally-Driven Buoyant Flow / Heat Pipes in EBS	- Vapor flow	2.2.10.10.0A
2.1.11.13	Thermal Effects on Chemistry and Microbial Activity in EBS		2.1.11.08.0A
2.1.11.14	Thermal Effects on Transport in EBS	- Thermal diffusion (Soret effect) - Thermal osmosis	2.1.11.10.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.1.12.00	1.12. GAS SOURCES AND EFFECTS		
2.1.12.01	Gas Generation in EBS	- Repository Pressurization - Mechanical Damage to EBS Components - He generation from waste from alpha decay - H_2 generation from waste package corrosion - CO_2, CH_4, and H_2S generation from microbial degradation - Vaporization of water	2.1.12.01.0A 2.1.12.02.0A 2.1.12.03.0A 2.1.12.04.0A
2.1.12.02	Effects of Gas on Flow Through the EBS	- Two-phase flow - Gas bubbles [see also Buoyant Flow/Heat Pipes in 2.1.11.12]	2.1.12.06.0A 2.1.12.07.0A
2.1.12.03	Gas Transport in EBS	- Gas phase transport - Gas phase release from EBS	2.1.12.07.0A 2.1.12.06.0A 2.2.10.10.0A
2.1.12.04	Gas Explosions in EBS	[see also Flammable Gas from Waste in 2.1.02.05]	2.1.12.08.0A
2.1.13.00	1.13. RADIATION EFFECTS		
2.1.13.01	Radiolysis - In Waste Package - In Backfill - In Tunnel	- Gas generation - Altered water chemistry	2.1.13.01.0A
2.1.13.02	Radiation Damage to EBS Components - Waste Form - Waste Package - Backfill - Other EBS Components	- Enhanced waste form degradation - Enhanced waste package degradation - Enhanced backfill degradation - Enhanced degradation of other EBS components (liner/rock reinforcement materials, seals, waste support structure)	2.1.13.02.0A
2.1.13.03	Radiological Mutation of Microbes		2.1.13.03.0A
2.1.14.00	1.14. NUCLEAR CRITICALITY		
2.1.14.01	Criticality In-Package	- Formation of critical configuration	2.1.14.15.0A 2.1.14.16.0A 2.1.14.21.0A 2.1.14.22.0A
2.1.14.02	Criticality in EBS or Near-Field	- Formation of critical configuration	2.1.14.17.0A 2.1.14.23.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.00.00	2. GEOLOGICAL ENVIRONMENT		
2.2.01.00	2.01. EXCAVATION DISTURBED ZONE (EDZ)		
2.2.01.01	Evolution of EDZ	- Lateral extent, heterogeneities - Physical properties - Flow pathways - Chemical characteristics of groundwater in EDZ - Radionuclide speciation and solubility in EDZ - Thermal-mechanical effects - Thermal-chemical alteration [see also Mechanical Effects of Excavation in 1.1.02.02]	2.2.01.04.0A
2.2.02.00	2.02. HOST ROCK		
2.2.02.01	Stratigraphy and Properties of Host Rock	- Rock units - Thickness, lateral extent, heterogeneities, discontinuities, contacts - Physical properties - Flow pathways [see also Fractures in 2.2.05.01 and Faults in 2.2.05.02]	2.2.03.01.0A 2.2.03.02.0A
2.2.03.00	2.03. OTHER GEOLOGIC UNITS		
2.2.03.01	Stratigraphy and Properties of Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Rock units - Thickness, lateral extent, heterogeneities, discontinuities, contacts - Physical properties - Flow pathways [see also Fractures in 2.2.05.01 and Faults in 2.2.05.02]	2.2.03.01.0A 2.2.03.02.0A
2.2.05.00	2.05. FLOW AND TRANSPORT PATHWAYS		
2.2.05.01	Fractures - Host Rock - Other Geologic Units	- Rock properties [see also Stratigraphy and Properties in 2.2.02.01 and 2.2.03.01]	1.2.02.01.0A 2.2.07.13.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.05.02	Faults - Host Rock - Other Geologic Units	- Rock properties [see also Stratigraphy and Properties in 2.2.02.01 and 2.2.03.01]	1.2.02.02.0A 2.2.07.13.0A
2.2.05.03	Alteration and Evolution of Geosphere Flow Pathways - Host Rock - Other Geologic Units	- Changes In rock properties - Changes in faults - Changes in fractures - Plugging of flow pathways - Changes in saturation [see also Stratigraphy and Properties in 2.2.02.01 and 2.2.03.01, Fractures in 2.2.05.01, and Faults in 2.2.05.02] [see also Thermal-Mechanical Effects in 2.2.11.06 and Thermal-Chemical Alteration in 2.2.11.07]	2.2.12.00.0A 2.2.12.00.0B
2.2.07.00	2.07. MECHANICAL PROCESSES		
2.2.07.01	Mechanical Effects on Host Rock	- From subsidence - From salt creep - From clay deformation - From granite deformation (rockfall / drift collapse into tunnels) - Chemical precipitation / dissolution - Stress regimes [see also Subsidence in 1.2.02.01, Thermal-Mechanical Effects in 2.2.11.06 and Thermal-Chemical Alteration in 2.2.11.07]	2.2.06.04.0A 2.2.06.05.0A
2.2.07.02	Mechanical Effects on Other Geologic Units	- From subsidence - Chemical precipitation / dissolution - Stress regimes [see also Subsidence in 1.2.02.01, Thermal-Mechanical Effects in 2.2.11.06 and Thermal-Chemical Alteration in 2.2.11.07]	2.2.06.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.08.00	2.08. HYDROLOGIC PROCESSES		
2.2.08.01	Flow Through the Host Rock	- Saturated flow - Fracture flow / matrix imbibition - Unsaturated flow (fingering, capillarity, episodicity, perched water) - Preferential flow pathways - Density effects on flow - Flow pathways out of Host Rock [see also Influx/Seepage into EBS in 2.1.08.09, Alteration of Flow Pathways in 2.2.05.03, Thermal Effects on Flow in 2.2.11.01, Effects of Gas on Flow in 2.2.12.02]	2.2.07.02.0A 2.2.07.03.0A 2.2.07.04.0A 2.2.07.05.0A 2.2.07.07.0A 2.2.07.08.0A 2.2.07.09.0A 2.2.07.12.0A
2.2.08.02	Flow Through the Other Geologic Units - Confining units - Aquifers	- Saturated flow - Fracture flow / matrix imbibition - Unsaturated flow (fingering, capillarity, episodicity, perched water) - Preferential flow pathways - Density effects on flow - Flow pathways out of Other Geologic Units [see also Alteration of Flow Pathways in 2.2.05.03, Thermal Effects on Flow in 2.2.11.01, Effects of Gas on Flow in 2.2.12.02]	2.2.07.02.0A 2.2.07.03.0A 2.2.07.04.0A 2.2.07.05.0A 2.2.07.07.0A 2.2.07.08.0A 2.2.07.09.0A 2.2.07.12.0A
2.2.08.03	Effects of Recharge on Geosphere Flow - Host Rock - Other Geologic Units	- Infiltration rate - Water table rise/decline [see also Infiltration in 2.3.08.03]	1.3.07.01.0A 1.3.07.02.0A 1.3.07.02.0B
2.2.08.04	Effects of Repository Excavation on Flow Through the Host Rock	- Saturated flow (flow sink) - Unsaturated flow (capillary diversion, drift shadow) - Influx/Seepage into EBS (film flow, enhanced seepage) [see also Influx/Seepage into EBS in 2.1.08.09]	2.1.08.02.0A 2.2.07.18.0A 2.2.07.20.0A 2.2.07.21.0A
2.2.08.05	Condensation Forms in Host Rock	- Condensation cap - Shedding [see also Thermal Effects on Flow in Geosphere in 2.2.11.01]	2.2.07.10.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.08.06	Flow Through EDZ	- Saturated / Unsaturated flow - Fracture / Matrix flow	2.2.01.03.0A
2.2.08.07	Mineralogic Dehydration	- Dehydration reactions release water and may lead to volume changes	2.2.10.14.0A
2.2.08.08	Groundwater Discharge to Biosphere Boundary	- Surface discharge (water table, capillary rise, surface water) - Flow across regulatory boundary	2.2.08.11.0A 2.3.11.04.0A
2.2.08.09	Groundwater Discharge to Well	- Human use (drinking water, bathing water, industrial) - Agricultural use (irrigation, animal watering)	1.4.07.02.0A
2.2.09.00	2.09.CHEMICAL PROCESSES - CHEMISTRY		
2.2.09.01	Chemical Characteristics of Groundwater in Host Rock	- Water composition (radionuclides, dissolved species, …) - Water chemistry (temperature, pH, Eh, ionic strength …) - Reduction-oxidation potential - Reaction kinetics - Interaction with EBS - Interaction with host rock [see also Chemistry in Tunnels in 2.1.09.04, Chemical Interactions and Evolution in 2.2.09.03] [contributes to Chemistry of Water Flowing into Repository in 2.1.09.01]	2.2.01.02.0B 2.2.08.01.0B
2.2.09.02	Chemical Characteristics of Groundwater in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Water composition (radionuclides, dissolved species, …) - Water chemistry (temperature, pH, Eh, ionic strength …) - Reduction-oxidation potential - Reaction kinetics - Interaction with other geologic units [see also Chemical Interactions and Evolution in 2.2.09.04]	2.2.08.01.0A
2.2.09.03	Chemical Interactions and Evolution of Groundwater in Host Rock	- Host rock composition and evolution (granite, clay, salt ...) - Evolution of water chemistry in host rock - Chemical effects on density - Interaction with EBS - Reaction kinetics - Mineral dissolution/precipitation - Redissolution of precipitates after dry-out [contributes to Chemistry in Host Rock in 2.2.09.01]	2.2.01.02.0B 2.2.07.14.0A 2.2.08.03.0B 2.2.08.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.09.04	Chemical Interactions and Evolution of Groundwater in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Host rock composition and evolution (granite, clay, salt ...) - Evolution of water chemistry in host rock - Chemical effects on density - Reaction kinetics - Mineral dissolution/precipitation - Recharge chemistry [contributes to Chemistry in Other Geologic Units in 2.2.09.02]	2.2.07.14.0A 2.2.08.03.0A
2.2.09.05	Radionuclide Speciation and Solubility in Host Rock	- Dissolved concentration limits [controlled by Chemistry in Host Rock in 2.2.09.01]	2.2.08.07.0B
2.2.09.06	Radionuclide Speciation and Solubility in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Dissolved concentration limits [controlled by Chemistry in Other Geologic Units in 2.2.09.02]	2.2.08.07.0A
2.2.09.50	2.09. CHEMICAL PROCESSES - TRANSPORT		
2.2.09.51	Advection of Dissolved Radionuclides in Host Rock	- Flow pathways and velocity - Advective properties (porosity, tortuosity) - Dispersion - Matrix diffusion - Saturation [see also Gas Phase Transport in 2.2.12.03]	2.2.07.15.0B 2.2.08.08.0B
2.2.09.52	Advection of Dissolved Radionuclides in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Flow pathways and velocity - Advective properties (porosity, tortuosity) - Dispersion - Matrix diffusion - Saturation [see also Gas Phase Transport in 2.2.12.03]	2.2.07.15.0A 2.2.08.08.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.09.53	Diffusion of Dissolved Radionuclides in Host Rock	- Gradients (concentration, chemical potential) - Diffusive properties (diffusion coefficients) - Flow pathways and velocity - Saturation	2.2.08.05.0A
2.2.09.54	Diffusion of Dissolved Radionuclides in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Gradients (concentration, chemical potential) - Diffusive properties (diffusion coefficients) - Flow pathways and velocity - Saturation	2.2.07.17.0A
2.2.09.55	Sorption of Dissolved Radionuclides in Host Rock	- Surface complexation properties - Flow pathways and velocity - Saturation [see also Chemistry in Host Rock in 2.2.09.01]	2.2.08.09.0B
2.2.09.56	Sorption of Dissolved Radionuclides in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Surface complexation properties - Flow pathways and velocity - Saturation [see also Chemistry in Host Rock in 2.2.09.01]	2.2.08.09.0A
2.2.09.57	Complexation in Host Rock	- Presence of organic complexants (humates, fulvates, carbonates, …) - Enhanced transport of radionuclides associated with organic complexants [see Radionuclide Speciation in 2.2.09.05 for inorganic complexation]	2.1.09.21.0C 2.2.08.06.0B
2.2.09.58	Complexation in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Presence of organic complexants (humates, fulvates, carbonates, …) - Enhanced transport of radionuclides associated with organic complexants [see Radionuclide Speciation in 2.2.09.06 for inorganic complexation]	2.1.09.21.0B 2.2.08.06.0A
2.2.09.59	Colloidal Transport in Host Rock	- Flow pathways and velocity - Saturation - Advection - Dispersion - Diffusion - Sorption - Colloid concentration	2.2.08.10.0B

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.09.60	Colloidal Transport in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Flow pathways and velocity - Saturation - Advection - Dispersion - Diffusion - Sorption - Colloid concentration	2.2.08.10.0A
2.2.09.61	Radionuclide Transport Through EDZ	- Advection - Dispersion - Diffusion - Sorption	2.2.01.05.0A
2.2.09.62	Dilution of Radionuclides in Groundwater - Host Rock - Other Geologic Units	- Mixing with uncontaminated groundwater - Mixing at withdrawal well [see also Groundwater Discharge to Well in 2.2.08.09]	2.2.07.16.0A
2.2.09.63	Dilution of Radionuclides with Stable Isotopes - Host Rock - Other Geologic Units	- Mixing with stable and/or naturally occurring isotopes of the same element	3.2.07.01.0A
2.2.09.64	Radionuclide Release from Host Rock - Dissolved - Colloidal - Gas Phase	- Spatial and temporal distribution of releases to the Other Geologic Units or to the Biosphere (due to varying flow pathways and velocities, varying transport properties) [contributions from Dissolved in 2.2.09.51/53/55, Colloidal in 2.2.09.59, Gas Phase in 2.2.12.03, EDZ in 2.2.09.61]	
2.2.09.65	Radionuclide Release from Other Geologic Units - Dissolved - Colloidal - Gas Phase	- Spatial and temporal distribution of releases to the Biosphere (due to varying flow pathways and velocities, varying transport properties) [see also Groundwater Discharge to Biosphere Boundary in 2.2.08.08, Groundwater Discharge to Well in 2.2.08.09, Recycling of Accumulated Radionuclides in 2.3.09.55] [contributions from Dissolved in 2.2.09.52/54/56, Colloidal in 2.2.09.60, Gas Phase in 2.2.12.03]	1.4.07.02.0A 2.2.08.11.0A 2.3.11.04.0A 2.3.13.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.10.00	2.10. BIOLOGICAL PROCESSES		
2.2.10.01	Microbial Activity in Host Rock	- Formation of complexants - Formation and stability of microbial colloids - Biodegradation - Bioaccumulation [see also Complexation in Host Rock in 2.2.09.57]	2.2.09.01.0B
2.2.10.02	Microbial Activity in Other Geologic Units (Non-Host-Rock) - Confining units - Aquifers	- Formation of complexants - Formation and stability of microbial colloids - Biodegradation - Bioaccumulation [see also Complexation in Other Geologic Units in 2.2.09.58]	2.2.09.01.0A
2.2.11.00	2.11. THERMAL PROCESSES		
2.2.11.01	Thermal Effects on Flow in Geosphere - Repository-Induced - Natural Geothermal	- Altered saturation / relative humidity (dry-out, resaturation) - Altered gradients, density, and/or flow pathways - Vapor flow - Condensation	1.2.06.00.0A 2.2.07.11.0A 2.2.10.01.0A 2.2.10.03.0A 2.2.10.03.0B 2.2.10.11.0A 2.2.10.12.0A 2.2.10.13.0A
2.2.11.02	Thermally-Driven Flow (Convection) in Geosphere	- Convection	2.2.10.02.0A
2.2.11.03	Thermally-Driven Buoyant Flow / Heat Pipes in Geosphere	- Vapor flow	2.2.10.10.0A
2.2.11.04	Thermal Effects on Chemistry and Microbial Activity in Geosphere	- Mineral precipitation / dissolution - Altered solubility [contributes to Chemistry in 2.2.09.01 and 2.2.09.02]	2.2.10.06.0A 2.2.10.08.0A
2.2.11.05	Thermal Effects on Transport in Geosphere	- Thermal diffusion (Soret effect) - Thermal osmosis	

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.2.11.06	Thermal-Mechanical Effects on Geosphere	- Thermal expansion / compression - Altered properties of fractures, faults, rock matrix	2.2.01.02.0A 2.2.10.04.0A 2.2.10.04.0B 2.2.10.05.0A
2.2.11.07	Thermal-Chemical Alteration of Geosphere	- Mineral precipitation / dissolution - Altered properties of fractures, faults, rock matrix - Alteration of minerals / volume changes - Formation of near-field chemically altered zone (rind)	2.1.09.12.0A 2.2.10.06.0A 2.2.10.07.0A 2.2.10.08.0A 2.2.10.09.0A
2.2.12.00	2.12. GAS SOURCES AND EFFECTS		
2.2.12.01	Gas Generation in Geosphere	- Degassing (clathrates, deep gases) - Microbial degradation of organics - Vaporization of water	2.2.11.01.0A 2.2.11.02.0A
2.2.12.02	Effects of Gas on Flow Through the Geosphere	- Altered gradients and/or flow pathways - Vapor/air flow - Two-phase flow - Gas bubbles [see also Buoyant Flow/Heat Pipes in 2.2.11.03]	2.2.10.11.0A 2.2.11.01.0A 2.2.11.02.0A
2.2.12.03	Gas Transport in Geosphere	- Gas phase transport - Gas phase release from Geosphere	2.2.11.03.0A
2.2.14.00	2.14. NUCLEAR CRITICALITY		
2.2.14.01	Criticality in Far-Field	- Formation of critical configuration	2.2.14.09.0A 2.2.14.11.0A
2.3.00.00	3. SURFACE ENVIRONMENT		
2.3.01.00	3.01. SURFACE CHARACTERISTICS		
2.3.01.01	Topography and Surface Morphology	- Recharge and discharge areas	2.3.01.00.0A
2.3.02.01	Surficial Soil Type	- Physical and chemical attributes	2.3.02.01.0A
2.3.04.01	Surface Water	- Lakes, rivers, springs - Dams, reservoirs, canals, pipelines - Coastal and marine features - Water management activities	1.4.07.01.0A 2.3.06.00.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.3.05.01	Biosphere Characteristics	- Climate - Soils - Flora and fauna - Microbes - Evolution of biosphere (natural, anthropogenic – e.g., acid rain) [see also Climate Change in 1.3.01.01, Surficial Soil Type in 2.3.02.01, Microbial Activity in 2.3.10.01]	2.3.13.01.0A
2.3.07.00	3.07. MECHANICAL PROCESSES		
2.3.07.01	Erosion	- Weathering - Denudation - Subsidence [see also Subsidence in 1.2.02.01, Periglacial Effects in 1.3.04.01, Glacial Effects in 1.3.05.01, Surface Runoff in 2.3.08.02, and Soil and Sediment Transport in 2.3.09.53]	1.2.07.01.0A 2.2.06.04.0A
2.3.07.02	Deposition	- Weathering	1.2.07.02.0A
2.3.07.03	Animal Intrusion into Repository		2.3.09.01.0A
2.3.08.00	3.08. HYDROLOGIC PROCESSES		
2.3.08.01	Precipitation	- Spatial and temporal distribution [see also Climate Change in 1.3.01.01] [contributes to Infiltration in 2.3.08.03]	2.3.11.01.0A
2.3.08.02	Surface Runoff and Evapotranspiration	- Runoff, impoundments, flooding, increased recharge - Evaporation - Condensation - Transpiration (root uptake) [see also Climate Change in 1.3.01.01, Erosion in 2.3.07.01] [contributes to Infiltration in 2.3.08.03]	2.3.11.02.0A 2.2.06.04.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.3.08.03	Infiltration and Recharge	- Spatial and temporal distribution - Effect on hydraulic gradient - Effect on water table elevation [see also Topography in 2.3.01.01, Surficial Soil Type in 2.3.02.01] [contributes to Effects of Recharge in 2.2.08.03]	2.3.11.03.0A
2.3.09.00	3.09. CHEMICAL PROCESSES - CHEMISTRY		
2.3.09.01	Chemical Characteristics of Soil and Surface Water	- Altered recharge chemistry (natural) - Altered recharge chemistry (anthropogenic – e.g., acid rain) [contributes to Chemical Evolution of Groundwater in 2.2.09.04]	1.4.01.03.0A 1.4.06.01.0A
2.3.09.02	Radionuclide Speciation and Solubility in Biosphere	- Dissolved concentration limits	2.2.08.07.0C
2.3.09.03	Radionuclide Alteration in Biosphere	- Altered physical and chemical properties - Isotopic dilution	2.3.13.02.0A 3.2.07.01.0A
2.3.09.50	3.09. CHEMICAL PROCESSES - TRANSPORT		
2.3.09.51	Atmospheric Transport Through Biosphere	- Radionuclide transport in air, gas, vapor, particulates, aerosols - Processes include: wind, plowing, degassing, precipitation	3.2.10.00.0A
2.3.09.52	Surface Water Transport Through Biosphere	- Radionuclide transport and mixing in surface water - Processes include: lake mixing, river flow, spring discharge, overland flow, irrigation, aeration, sedimentation, dilution [see also Surface Water in 2.3.04.01]	2.3.04.01.0A
2.3.09.53	Soil and Sediment Transport Through Biosphere	- Radionuclide transport in or on soil and sediments - Processes include: fluvial (runoff, river flow), eolian (wind), saltation, glaciation, bioturbation (animals) [see also Erosion in 2.3.07.01, Deposition in 2.3.07.02]	2.3.02.03.0A 2.3.09.01.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
2.3.09.54	Radionuclide Accumulation in Soils	- Leaching/evaporation from discharge (well, groundwater upwelling) - Deposition from atmosphere or water (irrigation, runoff)	2.3.02.02.0A
2.3.09.55	Recycling of Accumulated Radionuclides from Soils to Groundwater	[see also Radionuclide Release in 2.2.09.65]	1.4.07.03.0A
2.3.10.00	3.10. BIOLOGICAL PROCESSES		
2.3.10.01	Microbial Activity in Biosphere	- Effect on biosphere characteristics - Effect on transport through biosphere	
2.3.11.00	3.11. THERMAL PROCESSES		
2.3.11.01	Effects of Repository Heat on Biosphere		2.3.13.03.0A
2.4.00.00	4. HUMAN BEHAVIOR		
2.4.01.00	4.01. HUMAN CHARACTERISTICS		
2.4.01.01	Human Characteristics	- Physiology - Metabolism - Adults, children [contributes to Radiological Toxicity in 3.3.06.02]	2.4.01.00.0A
2.4.01.02	Human Evolution	- Changing human characteristics - Sensitization to radiation - Changing lifestyle	1.5.02.00.0A 3.3.06.02.0A
2.4.04.00	4.04. LIFESTYLE		
2.4.04.01	Human Lifestyle	- Diet and fluid intake (food, water, tobacco/drugs, etc.) - Dwellings - Household activities - Leisure activities [see also Land and Water Use in 2.4.08.01] [contributes to Ingestion in 3.3.04.01, Inhalation in 3.3.04.02, External Exposure in 3.3.04.03]	2.4.04.01.0A 2.4.07.00.0A
2.4.08.00	4.08. LAND AND WATER USE		
2.4.08.01	Land and Water Use	- Agricultural (irrigation, plowing, fertilization, crop storage, greenhouses, hydroponics) - Farms and Fisheries (feed, water, soil) - Urban / Industrial (development, energy production, earthworks, population density) - Natural / Wild (grasslands, forests, bush, surface water)	2.4.08.00.0A 2.4.09.01.0B 2.4.09.02.0A 2.4.10.00.0A
2.4.08.02	Evolution of Land and Water Use	- New practices (agricultural, farming, fisheries) - Technological developments - Social developments (new/expanded communities)	1.4.08.00.0A 1.4.09.00.0A 2.4.09.01.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
3.0.00.00	3. RADIONUCLIDE / CONTAMINANT FACTORS (BIOSPHERE)		
3.1.00.00	1. CONTAMINANT CHARACTERISTICS		
3.2.00.00	2. RELEASE / MIGRATION FACTORS		
3.3.00.00	3. EXPOSURE FACTORS		
3.3.01.00	3.01. RADIONUCLIDE / CONTAMINANT CONCENTRATIONS		
3.3.01.01	Radionuclides in Biosphere Media	- Soil - Surface Water - Air - Plant Uptake - Animal (Livestock, Fish) Uptake - Bioaccumulation [contributions from Radionuclide Release from Geologic Units in 2.2.09.65, Transport Through Biosphere in 2.3.09.51/52/53/54/55]	3.3.02.01.0A 3.3.02.02.0A 3.3.02.03.0A
3.3.01.02	Radionuclides in Food Products	- Diet and fluid sources (location, degree of contamination, dilution with uncontaminated sources) - Foodstuff and fluid processing and preparation (water filtration, cooking techniques) [see also Land and Water Use in 2.4.08.01, Radionuclides in Biosphere Media in 3.3.01.01]	3.3.01.00.0A
3.3.01.03	Radionuclides in Non-Food Products	- Dwellings (location, building materials and sources, fuel sources) - Household products (clothing and sources, furniture and sources, tobacco, pets) - Biosphere media [see also Land and Water Use in 2.4.08.01, Radionuclides in Biosphere Media in 3.3.01.01]	3.3.03.01.0A
3.3.04.00	3.04. EXPOSURE MODES		
3.3.04.01	Ingestion	- Food products - Soil, surface water	3.3.04.01.0A
3.3.04.02	Inhalation	- Gases and vapors - Suspended particulates (dust, smoke, pollen)	3.3.04.02.0A
3.3.04.03	External Exposure	- Non-Food products - Soil, surface water	3.3.04.03.0A

Table D-2. Comprehensive List of Used Fuel Disposition Campaign FEPs Potentially Relevant to Generic Disposal Options (continued)

UFD FEP Number	Description	Associated Processes	Related FEPs (SNL 2008b)
3.3.06.00	3.06. TOXICITY / EFFECTS		
3.3.06.01	Radiation Doses	- Exposure rates (ingestion, inhalation, external exposure) - Dose conversion factors - Gases and vapors - Suspended particulates (dust, smoke, pollen)	3.3.05.01.0A 3.3.08.00.0A
3.3.06.02	Radiological Toxicity and Effects	- Human health effects from radiation doses	3.3.06.00.0A
3.3.06.03	Non-Radiological Toxicity and Effects	- Human health effects from non-radiological toxicity	3.3.07.00.0A

Appendix E

Safety Assessment Model Parameter Inputs

Appendix E—Safety Assessment Model Parameter Inputs

E-1. Inventory

The potential future used nuclear fuel and high-level radioactive waste inventory requiring disposal is estimated by Carter and Luptak (2010). It is assumed that the future used nuclear fuel and high-level radioactive waste inventory will be disposed of in more than one repository. For the three mined disposal options considered in this safety case report, a 70,000 MTHM capacity is assumed. For deep borehole disposal, the repository capacity affects the total number of boreholes required, but does not affect the conceptualization of an individual borehole.

The potential used nuclear fuel inventory under four future nuclear power generation scenarios is estimated by Carter and Luptak (2010, Section 3.2). The lowest estimate, 140,000 MTU of UNF in 2055, derives from the scenario that assumes all existing nuclear reactors will be decommissioned after 60 years of operation and will not be replaced with new reactor capacity. Under this future scenario a single pressurized water reactor assembly is assumed to contain 0.435 MTU (91,000 MTU/209,000 pressurized water reactor assemblies) (Carter and Luptak (2010, Table 3-5). The same 36 radionuclides (listed in Clayton et al. (2011, Table 3.1-8)) are assumed to represent the UNF inventory for each of the four disposal options. The initial mass of each radionuclide in a single pressurized water reactor assembly (reported as g/MTHM), assumes fuel with a burn-up of 60 GWd/MTHM and 4.73% enrichment aged 30 years after discharge from a reactor (Carter and Luptak 2010, Table C-1).

For the single-repository safety assessments of the three mined disposal options supporting the safety case, the UNF inventory was assumed to be contained in 16,000 waste packages, with each waste package containing 10 pressurized water reactor assemblies. This results in a single-repository inventory of 69,665 MTHM. For the deep borehole disposal simulations the UNF inventory in a single borehole was assumed to be contained in 400 waste packages, with each waste package containing 1 pressurized water reactor assembly. This results in a single-borehole inventory of 174 MTHM. Under these assumptions, approximately 400 boreholes would be required to dispose of 70,000 MTHM.

A few of the radionuclide half-lives were revised from the values used in the FY 2011 GDS models (Clayton et al. 2011, Table 3.1-3). The only significant change is for ^{129}I, which changed from 1.7×10^{7} yr to 1.57×10^{7} yr.

E-2. Deterministic Salt GDS Model

Values for input parameters for the deterministic salt GDS model supporting the safety case derive from the FY 2011 salt GDS model (Clayton et al. 2011, Section 3.1) [*GDSE Salt FY11 Baseline v2 (Ref Scenario May09-2011).gsm*]. Key changes include:

- Deterministic simulation with mean values for uncertain parameters (see Table E-1 for further details)
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Reduced repository length from 3,270 m to 2,146 m to be consistent with the smaller number of waste packages
- Brine flow rates through the near-field salt DRZ and far field interbed assumed to remain constant at the 1,000,000-year value until 10,000,000 years.

Deterministic values for uncertain parameters were calculated based on mean values of each uncertainty distribution. These deterministic values are summarized in Table E-1. Constant parameter values that are unchanged from their deterministic treatment in the FY 2011 salt GDS model are not listed in Table E-1.

Table E-1. Summary of Deterministic Approximations for the Salt GDS Model

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Waste Form			
UNF fractional degradation rate (yr^{-1})	Log Triangular	1×10^{-8} (min); 1×10^{-7} (mode); 1×10^{-6} (max)	1.53×10^{-5}
Near-Field Salt DRZ			
Am solubility (mol/L)	Triangular	1.85×10^{-7} (min); 5.85×10^{-7} (mode); 1.85×10^{-6} (max)	8.73×10^{-7}
Np solubility (mol/L)	Triangular	4.79×10^{-10} (min); 1.51×10^{-9} (mode); 4.79×10^{-9} (max)	2.26×10^{-9}
Pu solubility (mol/L)	Triangular	1.40×10^{-6} (min); 4.62×10^{-6} (mode); 1.53×10^{-5} (max)	7.11×10^{-6}
Tc solubility (mol/L)	Log Triangular	4.56×10^{-10} (min); 1.33×10^{-8} (mode); 3.91×10^{-7} (max)	3.17×10^{-8}
Th solubility (mol/L)	Triangular	2.00×10^{-3} (min); 4.00×10^{-3} (mode); $7.97\times\times10^{-3}$ (max)	4.66×10^{-3}
Sn solubility (mol/L)	Triangular	9.87×10^{-9} (min); $2.66x10^{-8}$ (mode); 7.15×10^{-8} (max)	3.60×10^{-8}
U solubility (mol/L)	Triangular	4.89×10^{-8} (min); 1.12×10^{-7} (mode); 2.57×10^{-7} (max)	1.39×10^{-7}
Waste Package (degraded) porosity	Uniform	0.30 (min); 0.50 (max)	0.40
Salt porosity	Log-uniform	0.010 (min); 0.100 (max)	0.039
Brine Flow Rate to Underlying Interbed (m/yr)	N/A	Sampled from 100 brine flow rate histories (Clayton et al. 2011, Section 3.1.3)	8.56×10^{-7}
Far-Field Interbed			
Am solubility (mol/L)	Triangular	3.34×10^{-7} (min); 1.06×10^{-6} (mode); 3.34×10^{-6} (max)	1.58×10^{-6}
Np solubility (mol/L)	Log Triangular	1.11×10^{-6} (min); 1.11×10^{-5} (mode); 1.11×10^{-4} (max)	1.70×10^{-5}
Pu solubility (mol/L)	Triangular	7.80×10^{-7} (min); 2.58×10^{-6} (mode); 8.55×10^{-6} (max)	3.97×10^{-6}
Th solubility (mol/L)	Triangular	8.84×10^{-6} (min); 1.76×10^{-5} (mode); 3.52×10^{-5} (max)	2.05×10^{-5}
Sn solubility (mol/L)	Triangular	1.78×10^{-8} (min); 4.80×10^{-8} (mode); 1.29×10^{-7} (max)	6.49×10^{-8}
U solubility (mol/L)	Triangular	9.16×10^{-5} (min); 2.64×10^{-4} (mode); 7.62×10^{-4} (max)	3.73×10^{-4}
Brine Flow Rate in Interbed (m/yr)	N/A	Sampled from 100 brine flow rate histories (Clayton et al. 2011, Section 3.1.3)	3.96×10^{-7}

Table E-1. Summary of Deterministic Approximations for the Salt GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Ac k_d (mL/g)	Log-uniform	5 (min); 500 (max)	107.5
Am k_d (mL/g)	Uniform	25 (min); 100 (max)	62.5
C k_d (mL/g)	Uniform	0 (min); 0.6 (max)	0.3
Cm k_d (mL/g)	Log-uniform	5 (min); 500 (max)	107.5
Cs k_d (mL/g)	Uniform	1 (min); 20 (max)	10.5
Np k_d (mL/g)	Uniform	1 (min); 10 (max)	5.5
Pu k_d (mL/g)	Uniform	70 (min); 100 (max)	85
Pa k_d (mL/g)	Log-uniform	1 (min); 500 (max)	80.3
Ra k_d (mL/g)	Uniform	1 (min); 80 (max)	40.5
Se k_d (mL/g)	Uniform	0.2 (min); 0.5 (max)	0.35
Sn k_d (mL/g)	Uniform	2 (min); 10 (max)	6
Sr k_d (mL/g)	Uniform	1 (min); 80 (max)	40.5
Tc k_d (mL/g)	Uniform	0 (min); 2 (max)	1
Th k_d (mL/g)	Uniform	100 (min); 1000 (max)	550
U k_d (mL/g)	Uniform	0.2 (min); 1 (max)	0.6
Zr k_d (mL/g)	Log-uniform	3 (min); 500 (max)	97.1

NOTE: UNF fractional degradation rate (yr^{-1}) of 1.53×10^{-5} is larger than the maximum of the distribution of values for the purpose of examining performance with a very conservative EBS.

E-3. Deterministic Clay GDS Model

Values for input parameters for the deterministic clay GDS model supporting the safety case derive from the FY 2011 clay GDS model (Clayton et al. 2011, Section 3.3) [*FY11_Clay_GDSE_Model_0105.gsm*]. Key changes include:

- Deterministic simulation with mean values for uncertain parameters (see Table E-2 for further details)
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Instantaneous waste package failure
- Clay thickness of 150 m overlying the emplaced waste, consistent with Hansen et al. (2010, Figure 2.1-1 and Section 4)
- Equivalent diffusive releases to the far-field clay in both the upward and downward directions

Deterministic values for uncertain parameters were calculated based on mean values of each uncertainty distribution. These deterministic values are summarized in Table E-2. Constant parameter values that are unchanged from their deterministic treatment in the FY 2011 clay GDS model are not listed in Table E-2.

Table E-2. Summary of Deterministic Approximations for the Clay GDS Model

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Waste Package			
Ac k_d (mL/g)	Log Triangular	1000 (min); 5000 (mode); 5000 (max)	3125
Am k_d (mL/g)	Log Triangular	1000 (min); 5000 (mode); 5000 (max)	3125
C k_d (mL/g)	Log Triangular	10 (min); 100 (mode); 100 (max)	52.9
Cs k_d (mL/g)	Log Triangular	0 (min); 300 (mode); 300 (max)	5.1
Np k_d (mL/g)	Log Triangular	500 (min); 1000 (mode); 1000 (max)	804
Pu k_d (mL/g)	Log Triangular	1000 (min); 5000 (mode); 5000 (max)	3125
Pa k_d (mL/g)	Log Triangular	500 (min); 1000 (mode); 1000 (max)	804
Ra k_d (mL/g)	Log Triangular	0 (min); 500 (mode); 500 (max)	8.5
Sr k_d (mL/g)	Log Triangular	0 (min); 20 (mode); 20 (max)	0.35
Th k_d (mL/g)	Log Triangular	1000 (min); 5000 (mode); 5000 (max)	3125
U k_d (mL/g)	Log Triangular	100 (min); 1000 (mode); 1000 (max)	529
Near-Field Bentonite Buffer			
Am solubility (mol/L)	Log Triangular	1.0×10^{-12} (min); 1.0×10^{-10} (mode); 1.0×10^{-8} (max)	4.6×10^{-10}
C solubility (mol/L)	Log Triangular	2.3×10^{-5} (min); 2.3×10^{-3} (mode); 2.3×10^{-1} (max)	1.1×10^{-2}
Np solubility (mol/L)	Log Triangular	4.0×10^{-11} (min); 4.0×10^{-9} (mode); 4.0×10^{-7} (max)	1.8×10^{-8}
Nb solubility (mol/L)	Log Triangular	2.0×10^{-9} (min); 2.0×10^{-7} (mode); 2.0×10^{-5} (max)	9.2×10^{-7}
Pd solubility (mol/L)	Log Triangular	4.0×10^{-9} (min); 4.0×10^{-7} (mode); 4.0×10^{-5} (max)	1.8×10^{-6}
Pu solubility (mol/L)	Log Triangular	1.99×10^{-9} (min); 1.99×10^{-7} (mode); 1.99×10^{-5} (max)	9.2×10^{-7}
Se solubility (mol/L)	Log Triangular	5.0×10^{-12} (min); 5.0×10^{-10} (mode); 5.0×10^{-8} (max)	2.3×10^{-9}
Tc solubility (mol/L)	Log Triangular	4.0×10^{-11} (min); 4.0×10^{-9} (mode); 4.0×10^{-7} (max)	1.8×10^{-8}
Th solubility (mol/L)	Log Triangular	1.0×10^{-11} (min); 1.0×10^{-9} (mode); 1.0×10^{-7} (max)	4.6×10^{-9}
Sn solubility (mol/L)	Log Triangular	1.0×10^{-10} (min); 1.0×10^{-8} (mode); 1.0×10^{-6} (max)	4.6×10^{-8}

Table E-2. Summary of Deterministic Approximations for the Clay GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
U solubility (mol/L)	Log Triangular	5.0×10^{-10} (min); 5.0×10^{-8} (mode); 5.0×10^{-6} (max)	2.3×10^{-7}
Zr solubility (mol/L)	Log Triangular	2.0×10^{-10} (min); 2.0×10^{-8} (mode); 2.0×10^{-6} (max)	9.2×10^{-8}
Bulk density (kg/m^3)	Triangular	2,070 (min); 2,300 (mode); 2,530 (max)	2,300
Tortuosity	Triangular	0.072 (min); 0.725 (mode); 1.0 (max)	0.599
Available porosity – anions (C, Cl, I, Nb, Se)	Triangular	0.001 (min); 0.01 (mode); 1.0 (max)	0.337
Available porosity – cations	Triangular	0.10 (min); 1.0 (mode); 1.0 (max)	0.700
Am k_d (mL/g)	Log Triangular	120 (min); 12,000 (mode); 1.2×10^{6} (max)	55,457
Cs k_d (mL/g)	Log Triangular	0.437 (min); 43.7 (mode); 4370 (max)	202
Nb k_d (mL/g)	Log Triangular	315 (min); 31,500 (mode); 3.15×10^{6} (max)	145,580
Np k_d (mL/g)	Log Triangular	10 (min); 1,000 (mode); 100,000 (max)	4622
Pd k_d (mL/g)	Log Triangular	3.94 (min); 394 (mode); 39,400 (max)	1821
Pu k_d (mL/g)	Log Triangular	10 (min); 1,000 (mode); 100,000 (max)	4622
Se k_d (mL/g)	Log Triangular	0.01 (min); 1 (mode); 100 (max)	4.6
Sn k_d (mL/g)	Log Triangular	48.1 (min); 4,810 (mode); 481,000 (max)	42,854
Tc k_d (mL/g)	Log Triangular	131 (min); 13,100 (mode); 1.31×10^{6} (max)	60,726
Th k_d (mL/g)	Log Triangular	30 (min); 3,000 (mode); 300,000 (max)	13,864
U k_d (mL/g)	Log Triangular	1000 (min); 100,000 (mode); 1×10^{7} (max)	462,150
Zr k_d (mL/g)	Log Triangular	437 (min); 43,700 (mode); 4.37×10^{6} (max)	389,300
Near-Field Clay DRZ			
Ac solubility (mol/L)	Log Triangular	4.0×10^{-9} (min); 4.0×10^{-7} (mode); 4.0×10^{-5} (max)	1.8×10^{-6}
Am solubility (mol/L)	Log Triangular	4.0×10^{-9} (min); 4.0×10^{-7} (mode); 4.0×10^{-5} (max)	1.8×10^{-6}
C solubility (mol/L)	Log Triangular	2.3×10^{-5} (min); 2.3×10^{-3} (mode); 2.3×10^{-1} (max)	1.1×10^{-2}
Cm solubility (mol/L)	Log Triangular	4.0×10^{-9} (min); 4.0×10^{-7} (mode); 4.0×10^{-5} (max)	1.8×10^{-6}
Np solubility (mol/L)	Log Triangular	4.0×10^{-11} (min); 4.0×10^{-9} (mode); 4.0×10^{-7} (max)	1.8×10^{-8}

Table E-2. Summary of Deterministic Approximations for the Clay GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Nb solubility (mol/L)	Log Triangular	2.0×10^{-9} (min); 2.0×10^{-7} (mode); 2.0×10^{-5} (max)	9.2×10^{-7}
Pa solubility (mol/L)	Log Triangular	1.0×10^{-8} (min); 1.0×10^{-6} (mode); 1.0×10^{-4} (max)	4.6×10^{-6}
Pb solubility (mol/L)	Log Triangular	4.0×10^{-8} (min); 4.0×10^{-6} (mode); 4.0×10^{-4} (max)	1.8×10^{-5}
Pd solubility (mol/L)	Log Triangular	4.0×10^{-9} (min); 4.0×10^{-7} (mode); 4.0×10^{-5} (max)	1.8×10^{-6}
Pu solubility (mol/L)	Log Triangular	2.0×10^{-9} (min); 2.0×10^{-7} (mode); 2.0×10^{-5} (max)	9.2×10^{-7}
Ra solubility (mol/L)	Log Triangular	1.0×10^{-9} (min); 1.0×10^{-7} (mode); 1.0×10^{-5} (max)	4.6×10^{-7}
Se solubility (mol/L)	Log Triangular	5.0×10^{-12} (min); 5.0×10^{-10} (mode); 5.0×10^{-8} (max)	2.3×10^{-9}
Tc solubility (mol/L)	Log Triangular	4.0×10^{-11} (min); 4.0×10^{-9} (mode); 4.0×10^{-7} (max)	1.8×10^{-8}
Th solubility (mol/L)	Log Triangular	6.0×10^{-9} (min); 6.0×10^{-7} (mode); 6.0×10^{-5} (max)	2.8×10^{-6}
Sn solubility (mol/L)	Log Triangular	1.0×10^{-10} (min); 1.0×10^{-8} (mode); 1.0×10^{-6} (max)	4.6×10^{-8}
U solubility (mol/L)	Log Triangular	7.0×10^{-9} (min); 7.0×10^{-7} (mode); 7.0×10^{-5} (max)	3.2×10^{-6}
Zr solubility (mol/L)	Log Triangular	2.0×10^{-10} (min); 2.0×10^{-8} (mode); 2.0×10^{-6} (max)	9.2×10^{-8}
Tortuosity	Triangular	0.060 (min); 0.60 (mode); 0.61 (max)	0.423
Available porosity – Anions (C, Cl, I, Nb, Se)	Triangular	0.002 (min); 0.02 (mode); 1.0 (max)	0.341
Available porosity – cations	Triangular	0.10 (min); 1.0 (mode); 1.0 (max)	0.700
Fracture spacing (m)	Triangular	0.25 (min); 0.50 (mode); 1.00 (max)	0.58
Fracture aperture (m)	Triangular	0.0005 (min); 0.0010 (mode); 0.0050 (max)	0.0022
Ac k_d (mL/g)	Log Triangular	500 (min); 50,000 (mode); 5.0×10^{6} (max)	231,070
Am k_d (mL/g)	Log Triangular	500 (min); 50,000 (mode); 5.0×10^{6} (max)	231,070
C k_d (mL/g)	Log Triangular	0.00414 (min); 0.414 (mode); 41.4 (max)	1.9
Cm k_d (mL/g)	Log Triangular	500 (min); 50,000 (mode); 5.0×10^{6} (max)	231,070
Cs k_d (mL/g)	Log Triangular	3.88 (min); 388 (mode); 38,800 (max)	1,849
Nb k_d (mL/g)	Log Triangular	48.1 (min); 4,810 (mode); 481,500 (max)	22,210

Table E-2. Summary of Deterministic Approximations for the Clay GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Np k_d (mL/g)	Log Triangular	9 (min); 900 (mode); 90,000 (max)	4,159
Pa k_d (mL/g)	Log Triangular	10 (min); 1,000 (mode); 100,000 (max)	4,622
Pb k_d (mL/g)	Log Triangular	1.6 (min); 160 (mode); 16,000 (max)	739
Pd k_d (mL/g)	Log Triangular	8.05 (min); 805 (mode); 80,500 (max)	3,722
Pu k_d (mL/g)	Log Triangular	9 (min); 900 (mode); 90,000 (max)	4,159
Ra k_d (mL/g)	Log Triangular	10 (min); 1,000 (mode); 100,000 (max)	4,622
Sn k_d (mL/g)	Log Triangular	161 (min); 16,100 (mode); 1.61×10^6 (max)	74,451
Tc k_d (mL/g)	Log Triangular	11.5 (min); 1,150 (mode); 115,000 (max)	5,324
Th k_d (mL/g)	Log Triangular	80 (min); 8,000 (mode); 800,000 (max)	36,972
U k_d (mL/g)	Log Triangular	80 (min); 8,000 (mode); 800,000 (max)	36,972
Zr k_d (mL/g)	Log Triangular	11.5 (min); 1,150 (mode); 115,000 (max)	5,324
Far-Field Clay Host Rock			
Solubility (mol/L)		same as Near-Field Clay DRZ	
Tortuosity		same as Near-Field Clay DRZ	
Available porosity		same as Near-Field Clay DRZ	
k_d (mL/g)		same as Near-Field Clay DRZ	

E-4. Deterministic Granite GDS Model

Values for input parameters for the deterministic granite GDS model supporting the safety case derive from the FY 2011 granite GDS model (Clayton et al. 2011, Section 3.2) [*generic_granite_undisturbed_36species+Dummy_FY11report.gsm*] and from a subsequent generic granite model [*Generic_PA_Model_R01_001v_Map_simplifiedGraniteGDS1.gsm*]. Key changes from the FY 2011 granite GDS model include:

- Deterministic simulation with mean values for uncertain parameters (see Table E-3 for further details)
- Waste inventory of 70,000 MTHM UNF in 16,000 waste packages
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Replace the three-dimensional representation of far-field fractured granite using the FEHM dynamically-linked library with a one-dimensional GoldSim pipe with matrix diffusion
- Replace the two-dimensional representation of bentonite buffer with a set of one-dimensional GoldSim cells.

- Update solubility values to be more representative of granite pore waters (based on Mariner et al. 2011, Table 2-5)
- Update distribution coefficients (k_d's) to be more representative of bentonite in the waste package and buffer, based on the waste package and bentonite k_d values used in the clay GDS model (Table E-2 and Clayton et al. 2011, Section 3.3.3.3)
- Update distribution coefficients (k_d's) to be more representative of granite in the host rock (based on Carbol and Engkvist 1997).
- Instantaneous failure of 1% (160) of the waste packages. This replaces the FY 2011 GDS model assumption that between 0.1% and 1% of the waste packages directly intersect a far-field fracture.

Deterministic values for uncertain parameters were calculated based on mean values of each uncertainty distribution. Table E-3 summarizes these deterministic values and also lists constant parameter values that changed. Constant parameter values that are unchanged from their deterministic treatment in the FY 2011 granite GDS model are not listed in Table E-3.

Table E-3. Summary of Deterministic Approximations for the Granite GDS Model

Parameter	**Distribution Type**	**FY 2011 GDS Model Probabilistic Values**	**Safety Case Deterministic Value**
Waste Form			
UNF fractional degradation rate (yr^{-1})	Log Triangular	1×10^{-8} (min); 1×10^{-7} (mode); 1×10^{-6} (max)	2×10^{-5}
Waste Package			
Ac solubility (mol/L)	Constant	Unlimited	6×10^{-6}
Am solubility (mol/L)	Triangular	1.85×10^{-7} (min); 5.85×10^{-7} (mode); 1.85×10^{-6} (max)	6×10^{-6}
Cm solubility (mol/L)	Constant	Unlimited	6×10^{-6}
Nb solubility (mol/L)	Constant	Unlimited	4×10^{-5}
Np solubility (mol/L)	Triangular	4.79×10^{-10} (min); 1.51×10^{-9} (mode); 4.79×10^{-9} (max)	1×10^{-9}
Pa solubility (mol/L)	Constant	Unlimited	1×10^{-9}
Pd solubility (mol/L)	Constant	Unlimited	3×10^{-6}
Pu solubility (mol/L)	Triangular	1.40×10^{-6} (min); 4.62×10^{-6} (mode); 1.53×10^{-5} (max)	2×10^{-7}
Ra solubility (mol/L)	Constant	Unlimited	1×10^{-6}
Sb solubility (mol/L)	Constant	Unlimited	1×10^{-7}
Se solubility (mol/L)	Constant	Unlimited	4×10^{-8}
Sn solubility (mol/L)	Triangular	9.87×10^{-9} (min); 2.66×10^{-8} (mode); 7.15×10^{-8} (max)	3×10^{-8}
Tc solubility (mol/L)	Log Triangular	4.56×10^{-10} (min); 1.33×10^{-8} (mode); 3.91×10^{-7} (max)	3×10^{-8}
Th solubility (mol/L)	Triangular	2.00×10^{-3} (min); 4.00×10^{-3} (mode); 7.97×10^{-3} (max)	4×10^{-7}
U solubility (mol/L)	Triangular	4.89×10^{-8} (min); 1.12×10^{-7} (mode); 2.57×10^{-7} (max)	4×10^{-10}
Zr solubility (mol/L)	Constant	Unlimited	2×10^{-8}
Waste Package (degraded) porosity	Uniform	0.30 (min); 0.50 (max)	0.40

Table E-3. Summary of Deterministic Approximations for the Granite Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Ac k_d (mL/g)	Uniform	300 (min); 29,400 (max)	3125
Am k_d (mL/g)	Uniform	300 (min); 29,400 (max)	3125
C k_d (mL/g)	Constant	5	52.9
Cm k_d (mL/g)	Uniform	300 (min); 29,400 (max)	0
Cs k_d (mL/g)	Uniform	120 (min); 1,000 (max)	4.9
I k_d (mL/g)	Uniform	0 (min); 13 (max)	0
Np k_d (mL/g)	Uniform	30 (min); 1,000 (max)	804
Pa k_d (mL/g)	Uniform	30 (min); 1,000 (max)	804
Pu k_d (mL/g)	Uniform	150 (min); 16,800 (max)	3125
Ra k_d (mL/g)	Uniform	50 (min); 3,000 (max)	8.2
Se k_d (mL/g)	Uniform	4 (min); 30 (max)	0
Sr k_d (mL/g)	Uniform	50 (min); 3,000 (max)	0.34
Tc k_d (mL/g)	Uniform	50,000 (min); 60,000 (max)	0
Th k_d (mL/g)	Uniform	63 (min); 23,500 (max)	3125
U k_d (mL/g)	Uniform	90 (min); 1,000 (max)	529
Near-Field Bentonite Buffer			
Bulk density (kg/m^3)	Constant	2780	1562
Porosity	Constant	0.18	0.435
Fraction connected to far-field fractures	Uniform	0.001 (min); 0.01 (max)	0.01
Ac k_d (mL/g)	Uniform	300 (min); 29,400 (max)	0
Am k_d (mL/g)	Uniform	300 (min); 29,400 (max)	55,458
C k_d (mL/g)	Constant	5	0
Cm k_d (mL/g)	Uniform	300 (min); 29,400 (max)	0
Cs k_d (mL/g)	Uniform	120 (min); 1,000 (max)	202
I k_d (mL/g)	Uniform	0 (min); 13 (max)	0
Nb k_d (mL/g)	Constant	0	145,576
Np k_d (mL/g)	Uniform	30 (min); 1,000 (max)	4622
Pa k_d (mL/g)	Uniform	30 (min); 1,000 (max)	0
Pd k_d (mL/g)	Constant	0	1821
Pu k_d (mL/g)	Uniform	150 (min); 16,800 (max)	4622
Ra k_d (mL/g)	Uniform	50 (min); 3,000 (max)	0
Se k_d (mL/g)	Uniform	4 (min); 30 (max)	4.6
Sn k_d (mL/g)	Constant	0	22,252
Sr k_d (mL/g)	Uniform	50 (min); 3,000 (max)	0
Tc k_d (mL/g)	Uniform	50,000 (min); 60,000 (max)	60,726
Th k_d (mL/g)	Uniform	63 (min); 23,500 (max)	13,864

Table E-3. Summary of Deterministic Approximations for the Granite Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
U k_d (mL/g)	Uniform	90 (min); 1,000 (max)	462,146
Zr k_d (mL/g)	Constant	0	202,142
Near-Field Granite DRZ			
Ac solubility (mol/L)	Constant	Unlimited	6×10^{-6}
Am solubility (mol/L)	Triangular	3.34×10^{-7} (min); 1.06×10^{-6} (mode); 3.34×10^{-6} (max)	6×10^{-6}
Cm solubility (mol/L)	Constant	Unlimited	6×10^{-6}
Nb solubility (mol/L)	Constant	Unlimited	4×10^{-5}
Np solubility (mol/L)	Log Triangular	1.11×10^{-6} (min); 1.11×10^{-5} (mode); 1.11×10^{-4} (max)	1×10^{-9}
Pa solubility (mol/L)	Constant	Unlimited	1×10^{-9}
Pd solubility (mol/L)	Constant	Unlimited	3×10^{-6}
Pu solubility (mol/L)	Triangular	7.80×10^{-7} (min); 2.58×10^{-6} (mode); 8.55×10^{-6} (max)	2×10^{-7}
Ra solubility (mol/L)	Constant	Unlimited	1×10^{-6}
Sb solubility (mol/L)	Constant	Unlimited	1×10^{-7}
Se solubility (mol/L)	Constant	Unlimited	4×10^{-8}
Sn solubility (mol/L)	Triangular	1.78×10^{-8} (min); 4.80×10^{-8} (mode); 1.29×10^{-7} (max)	3×10^{-8}
Tc solubility (mol/L)	Constant	Unlimited	3×10^{-8}
Th solubility (mol/L)	Triangular	8.84×10^{-6} (min); 1.76×10^{-5} (mode); 3.52×10^{-5} (max)	4×10^{-7}
U solubility (mol/L)	Triangular	9.16×10^{-5} (min); 2.64×10^{-4} (mode); 7.62×10^{-4} (max)	4×10^{-10}
Zr solubility (mol/L)	Constant	Unlimited	2×10^{-8}
Porosity	Uniform	0.0005 (min); 0.01 (max)	0.0018
Tortuosity	Normal	0.0144 (mean); 4.176×10^{-3} (sdev)	0.011
Volumetric flow rate (m^3/yr)	Constant	4.5×10^{-4}	5.1×10^{-4}
Ac k_d (mL/g)	Uniform	300 (min); 29,400 (max)	2,485
Am k_d (mL/g)	Uniform	300 (min); 29,400 (max)	2,485
C k_d (mL/g)	Constant	5	1.1
Cm k_d (mL/g)	Uniform	300 (min); 29,400 (max)	2,485
Cs k_d (mL/g)	Uniform	120 (min); 1,000 (max)	39.1
I k_d (mL/g)	Uniform	0 (min); 13 (max)	0
Nb k_d (mL/g)	Constant	0	1,395
Np k_d (mL/g)	Uniform	30 (min); 1,000 (max)	3,909
Pa k_d (mL/g)	Uniform	30 (min); 1,000 (max)	1,954
Pd k_d (mL/g)	Constant	0	12.5
Pu k_d (mL/g)	Uniform	150 (min); 16,800 (max)	3,909
Ra k_d (mL/g)	Uniform	50 (min); 3,000 (max)	39.1
Se k_d (mL/g)	Uniform	4 (min); 30 (max)	2.0
Sn k_d (mL/g)	Constant	0	0.16

Table E-3. Summary of Deterministic Approximations for the Granite Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Sr k_d (mL/g)	Uniform	50 (min); 3,000 (max)	0.39
Tc k_d (mL/g)	Uniform	50,000 (min); 60,000 (max)	1,173
Th k_d (mL/g)	Uniform	63 (min); 23,500 (max)	3,909
U k_d (mL/g)	Uniform	90 (min); 1,000 (max)	3,909
Zr k_d (mL/g)	Constant	0	1,395
Far-Field Fractured Granite Host Rock			
Porosity	Uniform	0.0005 (min); 0.01 (max)	0.0018
Tortuosity	Normal	0.0144 (mean); 4.176×10^{-3} (sdev)	0.011
Fracture aperture (m)	Uniform	0.00001 (min); 0.00050 (max)	0.0002
Fracture spacing (m)	Constant	25	25
Fracture height (m)	Constant	1.00	3.12
Ac k_d (mL/g)	Cumulative	1,000 (min); 3,000 (mode); 5,000 (max)	2,485
Am k_d (mL/g)	Cumulative	1,000 (min); 3,000 (mode); 5,000 (max)	2,485
C k_d (mL/g)	Cumulative	0.5 (min); 1.0 (mode); 2.0 (max)	1.1
Cm k_d (mL/g)	Cumulative	1,000 (min); 3,000 (mode); 5,000 (max)	2,485
Cs k_d (mL/g)	Cumulative	100 (min); 500 (mode); 1000 (max)	39.1
Nb k_d (mL/g)	Cumulative	500 (min); 1,000 (mode); 3,000 (max)	1,395
Np k_d (mL/g)	Cumulative	1,000 (min); 5,000 (mode); 10,000 (max)	3,909
Pa k_d (mL/g)	Cumulative	500 (min); 1,000 (mode); 5,000 (max)	1,954
Pd k_d (mL/g)	Cumulative	10 (min); 100 (mode); 500 (max)	12.5
Pu k_d (mL/g)	Cumulative	1,000 (min); 5,000 (mode); 10,000 (max)	3,909
Ra k_d (mL/g)	Cumulative	50 (min); 100 (mode); 500 (max)	39.1
Se k_d (mL/g)	Cumulative	0.5 (min); 1.0 (mode); 5.0 (max)	2.0
Sn k_d (mL/g)	Cumulative	0 (min); 1.0 (mode); 10 (max)	0.16
Sr k_d (mL/g)	Cumulative	5 (min); 10 (mode); 50 (max)	0.39
Tc k_d (mL/g)	Cumulative	300 (min); 1,000 (mode); 3,000 (max)	1,173
Th k_d (mL/g)	Cumulative	1,000 (min); 5,000 (mode); 10,000 (max)	3,909
U k_d (mL/g)	Cumulative	1,000 (min); 5,000 (mode); 10,000 (max)	3,909
Zr k_d (mL/g)	Cumulative	500 (min); 1,000 (mode); 3,000 (max)	1,395

NOTE: UNF fractional degradation rate (yr^{-1}) of 2×10^{-5} is larger than the maximum of the distribution of values for the purpose of examining performance with a very conservative EBS. Some other deterministic values are outside the range of the probabilistic distribution of values due to updated property values considered more representative of a granite repository system.

E-5. Deterministic Deep Borehole GDS Model

Values for input parameters for the deterministic deep borehole GDS model supporting the safety case derive from the FY 2011 deep borehole GDS model (Clayton et al. 2011, Section 3.4) [*(UNF Base Perm_May26) DBH FY11 (Baseline v3_May23-2011).gsm*]. Key changes include:

- Deterministic simulation with mean values for uncertain parameters (see Table E-4 for further details)
- Waste inventory of 174 MTHM UNF per borehole in 400 waste packages
- Fractional waste form degradation rate of 2×10^{-5} yr^{-1}
- Fluid flow rates up the borehole assumed to remain constant at the 1,000,000-year values until 10,000,000 years

Deterministic values for uncertain parameters were calculated based on mean values of each uncertainty distribution. These deterministic values are summarized in Table E-4. Constant parameter values that are unchanged from their deterministic treatment in the FY 2011 deep borehole GDS model are not listed in Table E-4.

Table E-4. Summary of Deterministic Approximations for the Deep Borehole GDS Model

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Waste Form			
UNF fractional degradation rate (yr^{-1})	Log Triangular	1×10^{-8} (min); 1×10^{-7} (mode); 1×10^{-6} (max)	1.53×10^{-5}
Waste Disposal Zone			
Am solubility (mol/L)	Triangular	7.8×10^{-10} (min); 6.5×10^{-9} (mode); 4.4×10^{-8} (max)	1.7×10^{-8}
Np solubility (mol/L)	Triangular	6.0×10^{-7} (min); 1.9×10^{-6} (mode); 6.0×10^{-6} (max)	2.8×10^{-6}
Pu solubility (mol/L)	Triangular	3.40×10^{-14} (min); 3.56×10^{-14} (mode); 3.73×10^{-13} (max)	1.48×10^{-13}
Tc solubility (mol/L)	Log Triangular	4.56×10^{-10} (min); 1.33×10^{-8} (mode); 3.91×10^{-7} (max)	3.17×10^{-8}
Th solubility (mol/L)	Triangular	1.7×10^{-8} (min); 3.4×10^{-8} (mode); 6.8×10^{-8} (max)	3.9×10^{-8}
Sn solubility (mol/L)	Triangular	9.87×10^{-9} (min); 2.66×10^{-8} (mode); 7.15×10^{-8} (max)	3.60×10^{-8}
U solubility (mol/L)	Triangular	4.17×10^{-13} (min); 9.46×10^{-13} (mode); 2.19×10^{-12} (max)	1.18×10^{-12}

Table E-4. Summary of Deterministic Approximations for the Deep Borehole GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Fluid Flow Rate (m/yr)	N/A	Sampled from 100 flow rate histories (Clayton et al. 2011, Section 3.4.1.3)	1 flow rate history
Ac k_d (mL/g)	Log-uniform	5 (min); 500 (max)	107.5
Am k_d (mL/g)	Log-uniform	5 (min); 500 (max)	107.5
C k_d (mL/g)	Uniform	0 (min); 0.6 (max)	0.3
Cm k_d (mL/g)	Log-uniform	5 (min); 500 (max)	107.5
Cs k_d (mL/g)	Uniform	5 (min); 40 (max)	22.5
Np k_d (mL/g)	Log-uniform	1 (min); 500 (max)	80.3
Pu k_d (mL/g)	Log-uniform	1 (min); 500 (max)	80.3
Pa k_d (mL/g)	Log-uniform	1 (min); 500 (max)	80.3
Ra k_d (mL/g)	Uniform	0.4 (min); 3 (max)	1.7
Se k_d (mL/g)	Uniform	0.2 (min); 0.5 (max)	0.35
Sn k_d (mL/g)	Uniform	2 (min); 10 (max)	6
Sr k_d (mL/g)	Uniform	0.4 (min); 3 (max)	1.7
Tc k_d (mL/g)	Log-uniform	0.00001 (min); 25 (max)	1.7
Th k_d (mL/g)	Log-uniform	3 (min); 500 (max)	97.1
U k_d (mL/g)	Log-uniform	0.4 (min); 500 (max)	70.1
Zr k_d (mL/g)	Log-uniform	3 (min); 500 (max)	97.15
Seal Zone			
Fluid Flow Rate (m/yr)	N/A	Sampled from 100 flow rate histories (Clayton et al. 2011, Section 3.4.1.3)	1 flow rate history
Ac k_d (mL/g)	Log-uniform	300 (min); 29,400 (max)	6,347
Am k_d (mL/g)	Log-uniform	300 (min); 29,400 (max)	6,347
Cm k_d (mL/g)	Log-uniform	300 (min); 29,400 (max)	6,347
Cs k_d (mL/g)	Log-uniform	120 (min); 1,000 (max)	415
Np k_d (mL/g)	Log-uniform	30 (min); 1,000 (max)	277
Pa k_d (mL/g)	Log-uniform	30 (min); 1,000 (max)	277
Pd k_d (mL/g)	Uniform	5 (min); 12 (max)	8.5
Pu k_d (mL/g)	Log-uniform	150 (min); 16,800 (max)	3,529
Ra k_d (mL/g)	Log-uniform	50 (min); 3,000 (max)	721
Se k_d (mL/g)	Uniform	4 (min); 20 (max)	12
Sn k_d (mL/g)	Uniform	17 (min); 50 (max)	33.5
Sr k_d (mL/g)	Log-uniform	50 (min); 3,000 (max)	721
Tc k_d (mL/g)	Log-uniform	0.0001 (min); 250 (max)	17
Th k_d (mL/g)	Log-uniform	63 (min); 23,500 (max)	3,958
U k_d (mL/g)	Log-uniform	90 (min); 1,000 (max)	378
Zr k_d (mL/g)	Log-uniform	100 (min); 5,000 (max)	1,253

Table E-4. Summary of Deterministic Approximations for the Deep Borehole GDS Model (continued)

Parameter	Distribution Type	FY 2011 GDS Model Probabilistic Values	Safety Case Deterministic Value
Upper Borehole Zone			
Volumetric Fluid Flow Rate (m^3/yr)	Constant	0.00235	0.00235
Ac k_d (mL/g)	Log-uniform	100 (min); 100,000 (max)	14,462
Am k_d (mL/g)	Log-uniform	100 (min); 100,000 (max)	14,462
C k_d (mL/g)	Log-uniform	0.0001 (min); 2,000 (max)	119
Cm k_d (mL/g)	Log-uniform	100 (min); 100,000 (max)	14,462
Cs k_d (mL/g)	Log-uniform	10 (min); 10,000 (max)	1,446
Np k_d (mL/g)	Log-uniform	10 (min); 1,000 (max)	215
Pa k_d (mL/g)	Log-uniform	10 (min); 1,000 (max)	215
Pd k_d (mL/g)	Uniform	4 (min); 100 (max)	52
Pu k_d (mL/g)	Log-uniform	300 (min); 100,000 (max)	17,163
Ra k_d (mL/g)	Log-uniform	5 (min); 3,000 (max)	468
Se k_d (mL/g)	Uniform	1 (min); 8 (max)	4.5
Sn k_d (mL/g)	Log-uniform	50 (min); 700 (max)	246
Sr k_d (mL/g)	Log-uniform	5 (min); 3,000 (max)	468
Tc k_d (mL/g)	Log-uniform	0.0001 (min); 1,000 (max)	62
Th k_d (mL/g)	Log-uniform	800 (min); 60,000 (max)	13,711
U k_d (mL/g)	Log-uniform	20 (min); 1,700 (max)	378
Zr k_d (mL/g)	Log-uniform	10 (min); 8,300 (max)	1,233

NOTE: UNF fractional degradation rate (yr^{-1}) of 1.53×10^{-5} is larger than the maximum of the distribution of values for the purpose of examining performance with a very conservative EBS.